Canadian Nuclear Safety Commission | Commission canadienne de sûreté nucléaire

Annual CNSC Staff Report for 2003 on the Safety Performance of the Canadian Nuclear Power Industry

INFO-0745

November 2004

Canada

ANNUAL CNSC STAFF REPORT FOR 2003 ON THE SAFETY PERFORMANCE OF THE CANADIAN NUCLEAR POWER INDUSTRY

INFO-0745

Published by the
Canadian Nuclear Safety Commission
November 2004

Annual CNSC Staff Report for 2003 on the Safety Performance of the Canadian Nuclear Power Industry
INFO-0745 Document

Published by the Canadian Nuclear Safety Commission

Cat. No. CC171-1/2003E
ISBN 0-662-37281-6

Le présent document est disponible en français sous le titre « Rapport annuel 2003 du personnel de la CCSN sur le rendement en matière de sûreté des centrales nucléaires au Canada »

Document availability

This document is available on the CNSC Web site at *www.nuclearsafety.gc.ca*. To order a print copy of the document in English or French, please contact:

Communications and Information Management Directorate
Canadian Nuclear Safety Commission
280 Slater St.
P.O. Box 1046, Station B
Ottawa, Ontario K1P 5S9
CANADA

Telephone: (613) 995-5894 or 1-800-668-5284 (Canada only)
Facsimile: (613) 995-2915
E-mail: publications@cnsc-ccsn.gc.ca

TABLE OF CONTENTS

SUMMARY

This report summarizes the Canadian Nuclear Safety Commission staff's assessment of the Canadian nuclear power industry's safety performance in 2003. The report describes the licensees' programs and implementation in nine safety areas. In addition to the assessment of the safety areas and programs for each station, the report makes comparisons between stations, shows year-to-year trends, and highlights significant issues that pertain to the industry at large.

Of the 22 CANDU reactors with operating licences issued by the *Commission*, 16 provided power to the electrical grid in 2003. This includes two units (Pickering A Unit 4 and Bruce A Unit 4) that were restarted late in the year following rehabilitation.

Staff's assessment of the safety areas concludes that the power reactor industry operated safely in 2003. No worker at any power reactor station or member of the public received a radiation dose in excess of the regulatory limits. Emissions from all plants were also below regulatory limits. Excellent programs continue to be noted across the industry in the areas of Emergency Preparedness and Safeguards. In general, licensees had appropriate organizations in place to manage and safely operate their plants. However, the Performance Assurance safety area continued to be a problematic one for some licensees. Progress was observed in the Quality Management, Human Factors and Training programs, but continues to be slow for some stations.

The licensees' grades for the nine safety areas for 2003 are provided in the following tables (in bold font). The first table shows the "program" portion of the grade, and the second table shows the "implementation" portion. In both tables, the grades from the two previous annual reports for the stations are also shown for comparison. Explanation of changes in grades can be found in the details of this report. Of particular note is the fact that a new expectation for the Radiation Protection safety area was introduced and it had an impact on the grades of some of the sites in 2003. The radiation protection program at the Darlington and Gentilly-2 sites will be evaluated against this new expectation in the future.

The August 14, 2003 blackout had an impact, to varying degrees, on the operations at power reactor sites in Ontario. Several design issues at Pickering B were identified as a result of a focused inspection following the blackout. The staff of the Canadian Nuclear Safety Commission will continue to assess those issues, including their impact on safety, in 2004.

The grades assigned for each program and safety area are based on the rating system described in Appendix C.

Trends of "Program" Grades from Annual Reports for the Nine Safety Areas

Safety Area	Year of Report	Bruce		Darlington	Pickering		Gentilly-2	Point Lepreau
		A	B		A	B		
Operating Performance	2001	-	B	B	-	B	B	B
	2002	B	B	B	B	B	B	B
	2003	**B**	**B**	**B**	**B**	**B**	**B**	**B**
Performance Assurance	2001	-	B	B	-	B	C	C
	2002	B	B	B	B	B	C	C
	2003	**B**	**B**	**B**	**B**	**B**	**C**	**C**
Design & Analysis	2001	-	B	B	B	B	B	B
	2002	B	B	B	B	B	B	B
	2003	**B**	**B**	**B**	**B**	**C**	**B**	**B**
Equipment Fitness for Service	2001	-	B	B	B	B	B	B
	2002	B	B	B	B	B	B	B
	2003	**B**	**B**	**B**	**B**	**B**	**B**	**B**
Emergency Preparedness	2001	A	A	A	A	A	A	B
	2002	A	A	A	A	A	A	A
	2003	**A**	**A**	**A**	**A**	**A**	**A**	**A**
Environmental Performance	2001	B	B	C	B	B	C	C
	2002	B	B	B	B	B	B	B
	2003	**B**	**B**	**B**	**B**	**B**	**B**	**B**
Radiation Protection	2001	A	A	A	A	A	A	A
	2002	A	A	A	A	A	A	A
	2003	**B**	**B**	**A**	**B**	**B**	**A**	**B**
Site Security	2001	B	B	B	B	B	B	B
	2002	B	B	B	B	B	B	B
	2003	**B**	**B**	**B**	**B**	**B**	**B**	**B**
Safeguards	2001	A	A	A	A	A	A	A
	2002	A	A	A	A	A	A	A
	2003	**A**	**A**	**A**	**A**	**A**	**A**	**A**

Program grades for 2003 that changed since the 2002 annual report are highlighted.

Legend:

A = Exceeds requirements	B = Meets requirements	C = Below requirements	D = Significantly below requirements	E = Unacceptable

Trends of "Implementation" Grades from Annual Reports for the Nine Safety Areas

Safety Area	Year of Report	Bruce A	Bruce B	Darlington	Pickering A	Pickering B	Gentilly-2	Point Lepreau
Operating Performance	2001	-	B	C	-	C	B	B
	2002	B	B	B	B	B	B	B
	2003	**B**	**B**	**B**	**B**	**C**	**B**	**B**
Performance Assurance	2001	-	C	C	-	C	C	C
	2002	C	C	C	B	C	C	C
	2003	**B**	**B**	**C**	**C**	**B**	**C**	**C**
Design & Analysis	2001	-	B	B	B	B	B	B
	2002	B	B	B	B	B	B	B
	2003	**B**	**B**	**B**	**B**	**C**	**B**	**B**
Equipment Fitness for Service	2001	-	B	B	B	B	B	B
	2002	C	B	B	B	B	B	B
	2003	**B**	**B**	**B**	**B**	**B**	**B**	**B**
Emergency Preparedness	2001	A	A	A	A	A	A	B
	2002	A	A	A	A	A	A	C
	2003	**A**	**A**	**A**	**A**	**A**	**A**	**C**
Environmental Performance	2001	A	A	A	A	A	A	A
	2002	B	B	B	B	B	B	B
	2003	**B**	**B**	**B**	**B**	**B**	**B**	**B**
Radiation Protection	2001	B	B	B	B	B	C	B
	2002	B	B	B	B	B	C	B
	2003	**B**	**B**	**B**	**B**	**B**	**C**	**B**
Site Security	2001	B	B	B	B	B	B	B
	2002	B	B	B	B	B	B	B
	2003	**B**	**B**	**B**	**B**	**B**	**B**	**B**
Safeguards	2001	A	A	A	A	A	A	A
	2002	A	A	A	A	A	A	A
	2003	**A**	**A**	**A**	**A**	**A**	**A**	**B**

Implementation grades for 2003 that changed since the 2002 annual report are highlighted.

Legend:

A = Exceeds requirements	B = Meets requirements	C = Below requirements	D = Significantly below requirements	E = Unacceptable

INTRODUCTION

To meet the legal requirements of the *Nuclear Safety and Control Act* (NSCA) and Regulations, licensees must implement programs which provide adequate provisions for the protection of the environment, the health and safety of persons, the maintenance of national security, and the measures required to implement Canada's international obligations.

This report summarizes the Canadian Nuclear Safety Commission (CNSC) staff's assessment of the safety performance of nuclear power plant licensees in the Canadian nuclear power industry in 2003. The assessment is aligned with the legal requirements of the NSCA and Regulations, as well as the conditions of operating licences and applicable standards. Licensee programs are grouped into nine safety areas and the design of the programs and their implementation or performance are assessed. General descriptions of the safety areas and their constituent programs, as well as some of the relevant requirements, are provided in Section 1 of this report. Section 1 also makes comparisons between stations, shows year-to-year trends, and highlights significant issues that pertain to the industry at large. Section 2 focuses on the individual power reactor sites and provides more details related to the assessment of the safety areas and programs, especially where programs or performance fell below CNSC staff expectations.

The conclusions in this report are supported by information gathered through inspections by CNSC staff, document and event reviews, and CNSC performance indicators (PIs). The reactors at Bruce A and Pickering A were not operational for most or all of 2003. For their assessment, CNSC staff evaluated activities related to the restart of the reactors at those sites. Staff also made observations on Bruce Power and Ontario Power Generation's (OPG) programs that are generic to the sites to help assess safety at Bruce A and Pickering A.

Changes in mandatory reporting were imposed on power reactor licensees in 2003 via licence conditions that incorporated Regulatory Standard S-99, "Reporting Requirements for Operating Nuclear Power Plants". Among the information that licensees are required to report is an environmental monitoring report due on April 30 of every year. Hence, the assessment of the information in the 2003 environmental monitoring reports was not available for inclusion in this report.

The PIs for 2003 and previous years are tabulated and discussed in Section 3. PI data are submitted to the CNSC on a quarterly basis by each licensee.

Some specialized and technical terms are defined in Appendix A and are italicized upon first use in the text. The acronyms used in this document are listed in Appendix B, and the grades assigned for each program and safety area are based on the rating system described in Appendix C.

Important events or developments at the licensed sites in 2003 were reported to the Commission via *Commission Member Documents* (CMDs) called 'Significant Development Reports' (SDRs). Appendix D, which is based on the SDRs, describes the significant developments relevant to power reactors in 2003 and follow-up activities.

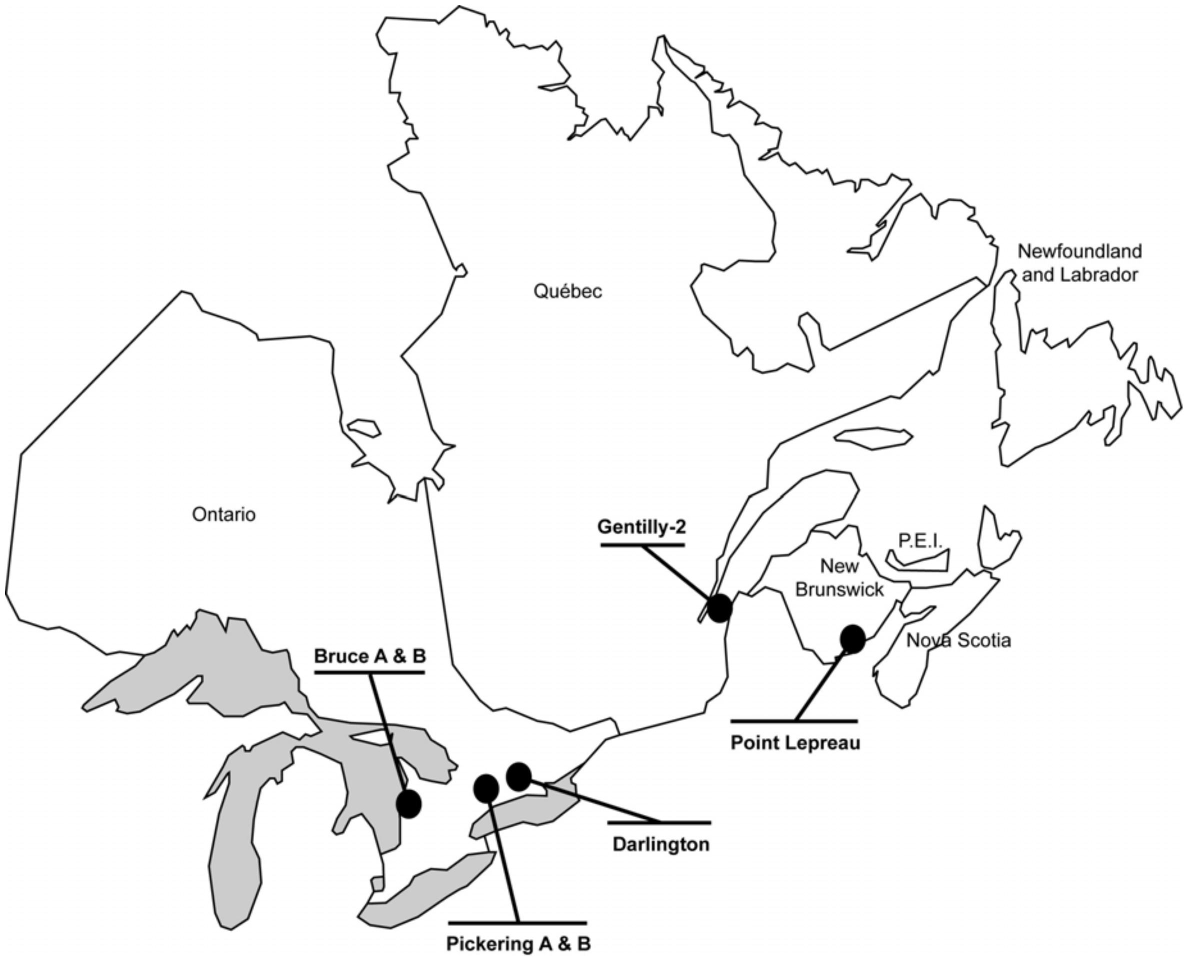
Finally, Appendix E describes the current status of the Generic Action Items related to each licensee.

SECTION 1

SAFETY PERFORMANCE ACROSS THE POWER REACTOR INDUSTRY

This Section of the report describes the safety areas and programs, and identifies some of the important requirements. CNSC staff rates licensee programs (P) and their implementation (I) separately, and the grades assigned for each program and safety area are based on the rating system as defined in Appendix C. Licensees' grades in 2003 for the safety areas and programs are also compared in this part of the report.

Figure 1 (on the following page) shows the location of all power reactor sites in Canada, the number and generating capacity of the reactors, their initial start-up date and licence holders, and expiry date of current licences. Of the 22 CANDU reactors with operating licences issued by the Commission, 16 provided power to the electrical grid in 2003. This includes two units (Pickering A Unit 4 and Bruce A Unit 4) that were restarted late in the year following rehabilitation work. Rehabilitation was conducted on four other reactors (Pickering A Units 1 to 3, and Bruce A Unit 3), while two reactors remained defuelled and in a *lay-up state* (Bruce A Units 1 and 2).



PLANT DATA							
Plant	Bruce A	Bruce B	Darlington	Pickering A	Pickering B	Gentilly- 2	Point Lepreau
Licensee	Bruce Power	Bruce Power	Ontario Power Generation	Ontario Power Generation	Ontario Power Generation	Hydro-Québec	New Brunswick Power
Reactor Units	4	4	4	4	4	1	1
Gross Electrical Capacity/Reactor (MW)	904	915	935	542	540	675	680
Start-Up	1976	1984	1989	1971	1982	1982	1982
Licence Expiry	2009/03/31	2009/03/31	2008/02/29	2005/06/30	2008/06/30	2006/12/31	2005/12/31

Figure 1: Locations and Data for Nuclear Power Plants in Canada

OPERATING PERFORMANCE

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	B	C	B	B	B	B

Operating Performance relates to station operation and organization and plant management. Organization and plant management is a cross-cutting program that takes into account findings from all safety areas that are applicable to the overall performance of the plant, such as safety culture and review of the transients. This safety area also includes non-radiological occupational health and safety.

The 14 reactors that were operational during most of the year (excluding Bruce A and Pickering A) were critical approximately 77% of the time. This number is lower than in 2002; the most noteworthy declines were at Pickering B, where the units were, on average, critical for 64% of the year and at Gentilly-2, where the reactor was critical for 65% of the year. The Bruce B reactors, when critical, were limited to 90% of full power.

Overall, CNSC staff reviews concluded that licensees generally had appropriate organizations in place to manage and operate their plants. The review of Operating Performance and the assessment of the other safety areas in this report support the conclusion that the Canadian power reactor industry was operated safely in 2003.

ORGANIZATION AND PLANT MANAGEMENT

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	B	C	B	B	B	B

Organization and Plant Management relates to the overall review of plant operation. This program covers high-level review topics and information from individual programs applicable to overall performance, as well as topics that fall under the direct responsibility of plant management. CNSC staff reviews concluded that appropriate organizations and programs were in place to safely operate the stations in 2003.

The sale of Bruce Power by the major shareholder, British Energy, to a new Canadian consortium was finalized in early 2003. CNSC staff observed no operational effects as a result of the sale. As part of its transition to a six unit organization, Bruce Power transferred the accountability for the maintenance division and for the work control and outage departments to the Chief Engineer. This is expected to foster better consistency and focus in maintenance-related work.

No worker at any station or member of the public received a radiation dose in excess of the regulatory limits in 2003. Emissions from all plants were also below regulatory limits. Low exposures and emissions continued to be areas of strength for the industry in 2003. These are general reflections of the contribution made to safety by effective organization and management at the stations. However, a degraded plant material condition in some systems at Pickering B, which led to a high number of transients and forced outages in 2003, was attributed to weaknesses in the implementation of the organizational and plant management systems. This is discussed in the following subsections, with further details provided in Section 2.

CNSC staff uses *action items* to bring issues to the attention of licensees that require corrective action to be taken in a timely manner. In 2003, CNSC staff opened 87 action items and closed 93. CNSC staff was satisfied with licensees' action item management, event reporting, plant system performance analysis and follow-up. In addition, CNSC staff continued to observe a low self-reporting threshold, indicative of a positive, questioning attitude of licensee staff.

A noteworthy development was the establishment of an event review panel by Bruce Power that meets monthly under the chairmanship of the chief executive officer. The main responsibility of the panel is to review all significant reportable events to ensure that the true root cause has been determined, and that effective corrective action has been taken.

OPERATIONS

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	B	B	B	B	B	B

The Operations program relates to the performance of the plant operating staff. It covers activities that operators perform to demonstrate safe operation of plant systems and awareness of the cool, control and contain philosophy. The program covers licensees' programs for operational inspections, procedural adherence, communications, approvals, change control and outage management. To verify these programs, CNSC staff carries out document reviews and field inspections of systems and operational practices. In 2003, CNSC staff completed approximately 65 Type I inspections (audits and evaluations) and 482 Type II inspections (equipment and system inspections and operating practice assessments). Most inspections confirmed compliance with CNSC requirements and the licensees' governing procedures and documents, and did not require any remedial action. For those inspections which did require remedial action, CNSC staff found that the licensees implemented appropriate measures to correct the deficiencies.

In 2003, there were 12 planned shutdowns for routine outages of the operating reactors (excluding Bruce A and Pickering A), lasting a total of 726 days. The longest outage was 130 days; this was the outage of Bruce B Unit 8 that lasted into 2004. CNSC staff monitors maintenance outages to ensure reactor safety principles are maintained, and to ensure that licensees' programs such as maintenance, radiation protection and dose control are effectively managed. CNSC staff reviews licensees' outage planning and organization to ensure that safety-significant work is completed. In 2003, CNCS staff found the planning and performance of these outages to be acceptable.

In summary, CNSC staff found that all licensees successfully implemented appropriate programs to demonstrate safe operation of plant systems.

OCCUPATIONAL HEALTH AND SAFETY (NON-RADIOLOGICAL)

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	B	B	B	B	B	B

Occupational Health and Safety is the program that both the employer and workers must implement to ensure that the risk posed by conventional hazards in the plant is minimized. All licensees met the requirements and expectations of this program and its implementation at all sites in 2003. At Gentilly-2, CNSC staff saw an improvement in the wearing of personal safety equipment (helmets, safety glasses). Staff has noted the presence of Hydro-Québec's management personnel in the field during numerous occasions, verifying compliance with occupational health and safety directives.

PERFORMANCE ASSURANCE

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	C	B	C	B	B	C	C	C	C

Performance Assurance relates to the organization's policies and programs and their impact on the level of quality and safety. Quality management, human performance and training are cross-cutting programs, meaning that performance in these programs affects performance in other programs as well as the effectiveness of overall plant management processes. CNSC staff rates this safety area through the assessment of the development, implementation and continuous improvement of policies, standards and procedures required to manage licensee programs. Performance Assurance groups three programs together: Quality Management, Human Factors and Training.

Weaknesses in Performance Assurance was noted for some of the sites in 2003. Grades of "C" for the safety area are based on two or more "C" grades for individual programs within the area. Darlington and Pickering A each received a "C" grade for implementation of programs under Performance Assurance, while Gentilly-2 and Point Lepreau each received a "C" grade for both the design and implementation of programs. The programs comprising the Performance Assurance safety area are described below, while details supporting the grades for individual sites are provided in Section 2.

QUALITY MANAGEMENT

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	C	B	C	B	C	C	C	C	C

Quality Management is the program of coordinated activities to direct and control an organization with regard to quality and safety. Quality management focuses on the achievement of results, in relation to the quality objectives, to satisfy the needs, expectations and requirements of interested parties as appropriate.

An operational Quality Management program requires the integrated series of processes documented in manuals, policies, standards and procedures to be those necessary for the safe operation of the plant. A licence condition for all plants specifies the Canadian Standards Association (CSA) N286 series of standards as the regulatory requirement for power reactor quality assurance programs.

In 2003, OPG continued to have problems with implementing effective corrective actions to programs that affect their pressure boundary program. This further delayed the issuance of the readiness report required by CNSC staff to schedule the implementation survey required for OPG to obtain the certificates of authorization for pressure boundary work. As a result, CNSC staff expanded the requirement for OPG to subcontract its modification/fabrication work on pressure boundaries to certified companies.

HUMAN FACTORS

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	C	B	B	B	B	C	C	C	C

Human Factors program relates to reducing human error by addressing factors that may affect human performance. The following are key human factors areas that CNSC staff review to ensure licensees' compliance with regulatory expectations: human factors in design, work organization and job design (e.g. staffing levels, hours of work), human performance in operating experience and *root-cause analysis*, human reliability, and usability aspects of procedures and job aids. In addition, licensees' human performance programs were added to the above list of Human Factors review topics in 2003.

CNSC staff continued to review activities in preparation for the refurbishment of the Point Lepreau station. The "C" ratings given to Gentilly-2 and Point Lepreau are due to weaknesses that still exist in their design change processes; details are in Section 2.

During 2003, a safety culture evaluation was conducted at Point Lepreau. Results indicated that the performance at Point Lepreau has improved since the organization and management evaluation that was conducted seven years ago. Data from the evaluation were compared with data collected at Point Lepreau through other inspection means. A workshop was also held to compare the results from the various data collection methods and to verify results with the analysis from the safety culture evaluation. Similarities and gaps were noted between the two collection methods; however, both methods can be integrated to produce a comprehensive profile of the organization. Work continues to allow CNSC staff to rate safety culture at each station.

CNSC site staff met with Darlington staff on a quarterly basis in support of the development of that licensee's safety culture program. OPG undertook several self-assessments and independent assessments of safety culture in 2003 using both internal and external peers.

A symposium on safety culture was held with the industry in early 2004. The delegates provided positive feedback to CNSC staff on this initiative. The CNSC has scheduled a follow-up workshop with industry representatives to continue the work begun at the symposium to arrive at a common understanding of this complex topic.

TRAINING

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	C	B	C	B	C	B	B	B	B	B	C

Training is the program that ensures that there exist adequate numbers of qualified staff to carry out the licensed activities. CNSC staff expects licensees to establish and implement adequate training programs to meet this requirement. These programs, including their testing methods, must provide licensee staff members in all relevant job areas with the knowledge and necessary skills to safely carry out their duties. Grades for Training are currently based on the review of training programs, using criteria based on the methodology called the *systematic approach to training* (SAT), and not the performance of licensee candidates in certification exams. However, ongoing satisfactory certification of workers is a requirement for all stations.

The focus of program evaluations in 2003 was on those certification training programs that were required in order to meet an acceptable standard prior to the transfer of the regulatory certification examinations to the licensees. So far, staff has conducted ten evaluations of station-specific training and simulator-based training for reactor operator and shift supervisor candidates at various plants. These evaluations identified deficiencies in most of the programs, and the licensees are currently addressing them through their corrective action plans. CNSC staff will conduct follow-up activities to verify the completion of the proposed corrective actions, and will continue with scheduled evaluations of the training programs. All licensees are graded "B" for the program portion of the rating since they have acceptable program descriptions and the necessary commitment. Even though the implementation status of the SAT is reflected in the "C" grades, it is noted that improvements were observed at all sites during the evaluation period. "C" grades are not abnormal at the early stage of such projects, and they are not indicative of deterioration in performance at those sites. Licensees are addressing the issues related to the transfer of exams.

CNSC staff conducts knowledge-based and performance-based examinations in order to assess the competence of licensee staff in safety-critical positions. During 2003, Phase I of the transfer of certification examinations from the CNSC to the licensees continued. In Phase I, overseen by CNSC staff, the licensees prepared, conducted and graded all written and simulator-based certification examinations for reactor operators and shift supervisor candidates. The licensees conducted this work in accordance with CNSC procedures. CNSC staff continued to approve and issue the certification examinations and the examination results.

In 2003, the success rate on the CNSC examinations for shift supervisor and control room operator candidates was 93% (142 of 152 candidates were successful). This was a decrease from the success rate of 96% (108 of 112) in 2002, but was still well above the average historical success rate of 86%.

The “Requirements for Requalification Testing of Certified Shift Personnel at Canadian Nuclear Power Plants” document was issued in July 2003 and is incorporated as a legal requirement via a licence condition for most sites, (the remaining licences will be amended in 2004). This document sets out the requirements for requalification tests that certified shift personnel at nuclear power plants must successfully complete in order for their certification to be renewed.

DESIGN AND ANALYSIS

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	C	C	B	B	B	B

The Design and Analysis safety area relates to the activities that impact on the ability of systems in a nuclear power plant to continually meet their design intent, given new information resulting from operating experience, safety analysis or the review of safety issues. When necessary, CNSC staff raises an action item with the licensee if a new failure or degradation mechanism is discovered. The licensee is then required to take interim compensatory measures to ensure that adequate safety margins of reactor operation are maintained. The issue is monitored until it has been satisfactorily and permanently resolved.

In 2003, CNSC staff reviews of Design and Analysis showed that all licensees continue to provide acceptable analysis and response to new safety issues. However, the performance of Pickering B in response to the August 14 blackout has raised some concerns regarding plant design of some systems. These issues contribute to the “C” grade assigned to Pickering B; more details are provided in section 2.

SAFETY ANALYSIS

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	B	B	B	B	B	B

Safety Analysis relates to the confirmation that the probability and consequences of a range of design basis events are acceptable. Analysis results also define safe operational limits. Power reactor licensees routinely carry out safety analyses to confirm that changes in the plant design are such that the consequences from design basis accidents continue to meet the requirements of the CNSC. CNSC staff reviews safety analyses mainly to verify that these analyses employ reasonably conservative assumptions, use validated models, have appropriate scope and demonstrate acceptable results.

A condition in the operating licence of power reactors requires the licensees to provide an update of the safety report to the CNSC every three years to ensure that the document continues to reflect the current facility design, operation and modifications to the safety analysis.

In 2003, CNSC staff reviews confirmed that licensees performed adequate safety analyses. In addition, all licensees submitted updates to their safety reports, as required.

SAFETY ISSUES

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	B	B	B	B	B	B

Safety Issues relates to the identification and resolution of issues arising from research, incorporation of new knowledge, hazard analysis or accident mitigation strategies.

A safety-related concern that cannot be resolved based on the currently available knowledge is referred to as an outstanding safety issue. CNSC staff has formally documented those outstanding safety issues that are common to more than one station and complex in nature as ‘generic action items’ (GAIs). Further work, occasionally including experimental research, is required to more accurately determine the overall effect of a GAI on the safety of the facility. Nevertheless, CNSC staff judges that continued station operation is permissible, because the majority of GAIs deal with situations where safety margins still exist but may be subject to potential degradation. Issues with confirmed, immediate safety significance are addressed by other means on a priority basis.

To ensure that CNSC expectations are clear for each GAI, CNSC staff has developed position statements that include closure criteria and an expected timeframe for closure. There has been progress on some issues in 2003, while progress on others proved to be slower than anticipated. Seventeen GAIs were active in 2003 and no new GAIs were created. Three GAIs were closed in

2003. Progress on each of the GAIs is described in Appendix E. CNSC staff is satisfied that adequate progress has been made on the remaining safety issues by all licensees.

DESIGN

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	C	C	B	B	B	B

Design relates to the upkeep of the initial plant specifications to align with modern standards, improved practices or correction of past deficiencies.

CNSC staff reviews plant design to ensure licensees maintain a documented description of equipment, including equipment qualification and classification requirements. CNSC staff reviews licensees' design change and safety enhancement programs, as well as programs that impact on the overall safe operation of the plant, such as fire protection. In 2003, CNSC staff continued to be satisfied with the industry's progress on physical changes made to the plants to resolve identified problems. Some noteworthy design changes are described in Section 2. CNSC staff is satisfied with licensees' progress in implementing physical upgrades for fire protection, initiated as a result of GAIs and a hazard analysis. The majority of the suppression and detection upgrade projects are now completed or are scheduled for completion in 2004.

Overall, the Design program at all sites met CNSC staff expectations, with the exception of Pickering B. Details supporting the grade for Pickering B, and its impact on the Design and Analysis safety area, are provided in Section 2.

EQUIPMENT FITNESS FOR SERVICE

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	B	B	B	B	B	B

Equipment Fitness for Service includes those programs that impact on the physical condition of structures, systems and components (SSC) in the plant. This safety area covers maintenance, structural integrity, reliability, and equipment qualification programs. To ensure that SSC important to safety in nuclear power plants are effective and remain so as the plant ages, licensees must establish adequate *environmental qualification* (EQ) programs and integrate the results of inspection and reliability programs into their plant maintenance activities.

CNSC staff reviews showed that the licensees generally met the expectations associated with Equipment Fitness for Service in 2003.

MAINTENANCE

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	C	C	B	B	B	B

Maintenance relates to the requirements and activities to maintain the plant systems, components and structures in a state that conforms to the current design requirements and analysis results.

Licensees are required to maintain their SSC in a state that conforms to the current design requirements and analysis results, and are required to implement a maintenance program that includes adequate organization, tools and procedures. Licensees must also demonstrate that related programs involving reliability, EQ, training, technical surveillance, procurement and planning effectively support this maintenance program.

The maintenance programs at the sites met staff's expectations, with the exception of Pickering B (see Section 2 for details).

STRUCTURAL INTEGRITY

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	B	B	B	B	B	B

Structural Integrity relates to the periodic inspections of major components to ensure that they remain fit for service.

CNSC staff requires that licensees establish strategies to manage structural integrity problems, including monitoring, assessing, mitigating, and, if appropriate, replacing degraded components. Licensees carry out periodic inspections to confirm that major primary heat transport system (HTS) and safety system components important to worker and public health and safety and the protection of the environment remain fit for service. The emphasis of these inspections is on *pressure tubes*, *feeder* piping and *steam generator* tubes.

In 2003, CNSC staff reviews found that licensees implemented adequate measures and appropriately adjusted their inspection programs to manage identified degradation. CNSC staff judged that, in 2003, licensees' equipment at all sites continued to be fit for service.

In 2003, CNSC staff directed the Bruce B, Gentilly-2 and Point Lepreau stations to update their existing periodic inspection programs to current standards. Gentilly-2 and Point Lepreau experienced problems with cracked feeders in 2003 that required repairs to be effected during shutdowns. More details on these issues are provided in Section 2.

RELIABILITY

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	B	B	B	B	B	B

Reliability relates to assessing, setting targets, testing, monitoring and reporting of plant systems whose failure impacts on the risk of a release of radioactive or hazardous material.

Licensees are required to ensure that systems whose failure impacts on the risk of a release of radioactive material be part of a reliability program. Licensees must establish a program that includes setting reliability targets, performing reliability assessments, testing and monitoring, and reporting. CNSC staff reviews of licensees' reliability programs mainly cover:

- reliability models and data verification;
- safety system availability;
- testing program; and
- reporting.

In 2003, all licensees completed their annual reliability reports, submitted their quarterly reports, continued to develop/update their reliability programs according to Regulatory Standard S-98, and continued to follow mandatory testing programs. All licensees met CNSC staff's expectations for their reliability programs. All safety systems met the regulatory targets for availability, although high pressure emergency core coolant (ECC) pumps and some safety support systems were unavailable for a few hours at Pickering B during the August 14 blackout. CNSC staff is currently assessing the impact of reliability of the electrical grid on reliability of safety equipment in the plants.

EQUIPMENT QUALIFICATION

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	B	B	B	B	B	B

Equipment Qualification relates to plant-specific functional and performance requirements which ensure that SSC are suitable for operation. An important part of the Equipment Qualification program is environmental qualification (EQ). The purpose of EQ is to ensure the capability of equipment to perform its intended safety function in an aged condition and under extreme environmental conditions resulting from design basis accidents. The CNSC requires, via licence conditions, that all nuclear power plant licensees have effective EQ programs implemented by June 30, 2004. To be deemed effective, the EQ programs must meet a number of acceptance criteria developed by CNSC staff. The licensees must:

a. have a documented EQ program and associated processes in place;
b. ensure that EQ processes and procedures meet recognized industry standards;

c. install (or replace) the required equipment and have evidence that it is qualified to perform its intended safety function;
d. have all EQ-related documentation available at the station;
e. develop a program to assess degradation and failures of qualified equipment during normal operation;
f. ensure that EQ-related processes comply with the station Quality Assurance (QA) program; and
g. train operations and maintenance staff on EQ principles and processes.

In recent years, licensees advanced their work on EQ programs and made good progress towards meeting the acceptance criteria above. CNSC staff reviewed and accepted licensees' EQ-related documentation that defines their EQ programs, processes and procedures. Item 'c' above represents the most important, resource-intensive part of the EQ upgrade work.

The progress at the sites toward meeting the EQ deadline is described in Section 2. The EQ upgrades vary considerably due to different EQ standards used at the time of design. For older stations designed when standards were not available, EQ was based on the use of high-grade electrical components of commercial quality. As a result, more equipment has been replaced at older plants than at plants qualified to more recent standards.

Other review topics under Equipment Qualification are chemistry control and fire protection. CNSC staff was satisfied both with the format and content of the summaries of chemistry performance submitted quarterly in 2003 under Regulatory Standard S-99. CNSC staff was generally satisfied with the licensees' programs for fire protection inspection as well as their implementation (with the exception of Pickering B, as detailed in Section 2).

EMERGENCY PREPAREDNESS

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
A	A	A	A	A	A	A	A	A	A	A	A	A	C

Emergency Preparedness relates to the consolidated Emergency Plan and the Emergency Preparedness Program, as well as the results of all Emergency Exercises.

To be able to respond effectively to an emergency, licensees must establish a consolidated emergency plan and an emergency preparedness program under that plan, and must ensure the response capability of their staff through simulated emergencies. To evaluate the emergency preparedness of a licensee, CNSC staff assesses the emergency plan and preparedness program, as well as the results of simulated emergency exercises. The assessment of the emergency plan provides an indication of the effectiveness of the emergency response strategy. The review of the emergency preparedness program verifies that all components of the emergency response plan are in place and maintained in a state of readiness. Finally, the evaluation of the facility's staff during a simulated nuclear accident provides an assessment of the emergency response capability.

In 2003, CNSC staff evaluated a full scale emergency exercise at Bruce A, a major exercise with the off-site authorities and organizations at Pickering B, a tabletop exercise (emergency exercise of limited scope) at Gentilly-2, and another major exercise with the off-site authorities and organizations at Point Lepreau. Ongoing compliance activities at other sites have found that licensees' programs and performance met or exceeded CNSC requirements. Generally, CNSC staff judges that emergency preparedness continues to be an industry strength. Details supporting the grades are provided in Section 2, including the "C" grade for implementation at Point Lepreau.

ENVIRONMENTAL PERFORMANCE

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	B	B	B	B	B	B

Environmental Performance relates to the programs to identify, control and monitor all releases of radioactive and hazardous substances from the facilities. This safety area includes radioactive and conventional waste management; effluent and environmental monitoring; emission data; unplanned releases; assessment of environmental protection systems, and compliance with federal and provincial environmental regulations.

CNSC regulations require that each licensee take all reasonable precautions to protect the environment and to control the release of radioactive and hazardous substances. CNSC staff verifies that licensees have programs in place to identify, control and monitor all releases of nuclear and hazardous substances from their plants. CNSC staff reviews of Environmental Performance include:

- public dose;
- emission data;
- effluent and environmental monitoring;
- unplanned releases; and
- assessment of environmental protection systems.

In 2003, data on airborne emissions and liquid releases of radioactive substances for all plants showed releases to the environment were consistently below the *derived release limits*. Doses to the public (in particular, members of the critical groups) were well below regulatory limits. As in previous years, these results demonstrate the continuance of a strong trend throughout the industry.

Licensees are required to report to the CNSC any unplanned releases of radioactive material or other hazardous substances to the environment. There were no significant reported unplanned releases of radioactive material or hazardous substances from any power reactor site in 2003.

RADIATION PROTECTION

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	A	B	B	B	B	B	A	C	B	B

Radiation Protection relates to the program in place to ensure protection of persons inside a nuclear facility from unnecessary exposure to ionizing radiation. The *Radiation Protection Regulations* prescribe dose limits for workers who may be exposed to radioactive material, and require that exposures to radiation be kept As Low As Reasonably Achievable (the ALARA principle).

CNSC staff carried out regular reviews of all aspects of radiation protection programs at all facilities and found that, in general, licensees continued to adequately manage radiation doses, as no worker received a radiation dose in excess of the regulatory limits. An exception was a shortcoming in the implementation of the Radiation Protection program at Gentilly-2; details are provided in Section 2.

A new requirement for the Radiation Protection safety area was introduced and it had an impact on the grades of some of the sites. In late 2002 and in 2003, CNSC staff evaluated the respiratory protection programs at Pickering, Bruce, and Point Lepreau. The evaluations introduced a new requirement that the respiratory protection programs conform to CSA Standard Z94.4-02 ("Selection, Care and Use of respirators") for the use of respirators to protect against radiological hazards. The evaluations determined that the licensee's documented programs did not cover, to varying degrees, radiological applications. For that reason, the program portion of the grade for Radiation Protection for those sites has been reduced from "A" to "B". CNSC staff is following up with the licensees to confirm that their respiratory protection programs are revised to meet the CSA standard. The radiation protection program at the Darlington and Gentilly-2 sites will be evaluated against this new requirement in the future; for now, the program portion of the grades for Darlington and Gentilly-2 remain "A".

SITE SECURITY

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
B	B	B	B	B	B	B	B	B	B	B	B	B	B

Site Security relates to the program required to implement and support the security requirements stipulated in the *Nuclear Security Regulations* and CNSC Order 01-1 for their facilities.

To obtain assurance of compliance with these requirements, CNSC staff assesses licensees':

- security guard service, including duties, responsibilities and training;
- nuclear response force, including equipment, training and deployment;
- protection arrangements with off-site response forces and testing of response plans;

- procedures to assess and respond to potential breaches of security; and
- security monitoring, assessment, detection, communication, access control systems, hardware and software.

Licensees are required to have a sufficient number of trained and properly-equipped security staff available at all times. Their sites must be continuously monitored and licensees must take appropriate action in the event of a security breach. In addition, while not directly specified by the regulations, CNSC staff expects all licensees to conduct joint security exercises with their respective off-site response forces.

In 2003, CNSC staff conducted a number of on-site security inspections and reviewed site-security reports. The results of these evaluations indicated that licensees were in compliance with applicable regulations as well as CNSC Order 01-1. In addition, CNSC staff assessed and approved approximately 200 applications for the import, export and transport of nuclear materials, all of which had security implications and proceeded without incident.

SAFEGUARDS

Bruce A		Bruce B		Darlington		Pickering A		Pickering B		Gentilly-2		Point Lepreau	
P	I	P	I	P	I	P	I	P	I	P	I	P	I
A	A	A	A	A	A	A	A	A	A	A	A	A	B

The CNSC's regulatory mandate includes ensuring conformity with measures required to implement Canada's international obligations under the Treaty on the Non-Proliferation of Nuclear Weapons. Pursuant to the Treaty, Canada has entered into a *safeguards* agreement with the *International Atomic Energy Agency* (IAEA). This agreement provides the IAEA with the right and the responsibility to verify that Canada is fulfilling its international commitment on the peaceful use of nuclear energy.

The CNSC provides the mechanism, through the NSCA and Regulations as well as licence conditions, for the IAEA to implement the safeguards agreement. Conditions for the application of IAEA safeguards are contained in power reactor operating licences, and compliance includes the timely provision of reports on the movement and location of all nuclear materials and measures for the application of IAEA safeguards.

In 2003, CNSC staff assessed licensees as either meeting or exceeding safeguards requirements. All reports required by the IAEA were provided in a timely manner. All licensees cooperated with the IAEA to successfully accomplish routine inspection activities, including design information verification, annual simultaneous physical inventory verification, complementary access and equipment installations. All licensees promptly addressed any problems or issues that arose, with one exception at Point Lepreau (discussed in Section 2).

SECTION 2

SAFETY PERFORMANCE AT THE POWER REACTOR SITES

This Section of the report is organized by power reactor site, with grades for safety areas and programs repeated from Section 1. The grades for the safety areas are also summarized in the two tables in the Summary beginning on page 1, where they are compared with grades from previous years.

Some noteworthy observations are made for each site in support of the grades, especially when the performance of licensees was below CNSC expectations.

BRUCE A AND B

OPERATING PERFORMANCE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Bruce A	OPERATING PERFORMANCE	B	B
	Organization & Plant Management	B	B
	Operations	B	B
	Occupational Health & Safety (Non-radiological)	B	B
Bruce B	OPERATING PERFORMANCE	B	B
	Organization & Plant Management	B	B
	Operations	B	B
	Occupational Health & Safety (Non-radiological)	B	B

With respect to the Operations program for Bruce B, CNSC staff observed no significant difficulties with the conduct of planned work during the outage of Bruce B Unit 8. The outage was extended (total length of 130 days) due to the discovery of degradation in steam generator tube support plates (see Section D.14). The assessment of radiation protection and conduct of planned maintenance during the outage of Bruce B Unit 5 met requirements and staff's expectations.

PERFORMANCE ASSURANCE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Bruce A	PERFORMANCE ASSURANCE	B	B
	Quality Management	B	B
	Human Factors	B	B
	Training	B	B
Bruce B	PERFORMANCE ASSURANCE	B	B
	Quality Management	B	B
	Human Factors	B	B
	Training	B	C

CNSC staff monitored inspection work and the completion of quality assurance (QA) activities related to the pre-commissioning and commissioning tasks involved with the return to service of Bruce A Units 3 and 4. The work was continuous but, as issues were identified, they were resolved daily or action items were raised to request a formal reply from Bruce Power. CNSC site staff issued 28 Type II inspection reports assessing compliance with the return-to-service project and the ongoing commissioning of Units 3 and 4. A CNSC inspection report covering the overall inspection was also prepared.

In November 2003, the CNSC inspected the supply chain and procurement assurance activities as part of the review of the licensee's management system manual. The inspection confirmed compliance with the applicable Canadian Standards Association (CSA) QA standard. Overall, CNSC staff finds the program and implementation associated with Bruce Power's Quality Management to be acceptable.

The Human Factors program was also assessed with respect to the restart activities at Bruce A. Bruce Power proposed changes to the minimum number of staff that would be present in the facility at all times, and CNSC staff required a safety-based justification prior to approving these changes. CNSC staff also reviewed the usability of selected procedures for dealing with abnormal incidents. An evaluation of the incorporation of human factors into modifications for restart was completed, and it was found that the human factors work completed met the expectations of CNSC staff.

In 2003, Bruce A completed the preparation of its training programs aimed at the restart of Units 3 and 4. The "C" grade for the Training program for Bruce B reflects the fact that the implementation status of the systematic approach to training (SAT) does not meet staff's expectations, although improvements were observed in 2003 and the licensee continues to address the issues related to the transfer of exams. The grade is not abnormal at this early stage of implementation.

DESIGN AND ANALYSIS

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Bruce A	DESIGN AND ANALYSIS	B	B
	Safety Analysis	B	B
	Safety Issues	B	B
	Design	B	B
Bruce B	DESIGN AND ANALYSIS	B	B
	Safety Analysis	B	B
	Safety Issues	B	B
	Design	B	B

The following noteworthy design changes were implemented at Bruce A in 2003 for the restart of Units 3 and 4 (Reference CMDs 03-H05 and 03-H27):

- seismic upgrades;
- fire protection upgrades;
- installation of qualified emergency power supply (diesel generators);
- installation of secondary control area;
- main control room steam proofing;
- emergency filtered air discharge system upgrades;
- installation of passive autocatalytic recombiners; and
- installation of powerhouse emergency ventilation system.

The following noteworthy design changes were implemented at Bruce B in 2003:

- installation of a design change for pressurizing the emergency core coolant (ECC) injection valve interspaces;

- rehabilitation of rectifiers and converters; and
- design modifications to augment the functionality of degraded portions of the top support plates in Unit 8 steam generators (see Section D.14).

EQUIPMENT FITNESS FOR SERVICE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Bruce A	EQUIPMENT FITNESS FOR SERVICE	B	B
	Maintenance	B	B
	Structural Integrity	B	B
	Reliability	B	B
	Equipment Qualification	B	B
Bruce B	EQUIPMENT FITNESS FOR SERVICE	B	B
	Maintenance	B	B
	Structural Integrity	B	B
	Reliability	B	B
	Equipment Qualification	B	B

Improved maintenance was partly credited for the safe return to service of Units 3 and 4 at Bruce A. The preventative and corrective maintenance backlogs at Bruce A remain high but are being managed. The preventative maintenance backlog at Bruce B remains high but is also being managed; the corrective maintenance backlog is acceptably low.

The major part of the remaining work to meet the environmental qualification (EQ) deadline at Bruce A involves updating documentation. CNSC staff's reviews showed that the fitness for service of Bruce A fire protection was acceptable.

The major part of the remaining work to meet the EQ deadline at Bruce B involves updating documentation. EQ-related work at the Bruce B vacuum building requires a vacuum building outage (i.e. all units); an outage is scheduled for the fall of 2004.

EMERGENCY PREPAREDNESS

Site	SAFETY AREA	Grades	
		Program	Implementation
Bruce A	EMERGENCY PREPAREDNESS	A	A
Bruce B	EMERGENCY PREPAREDNESS	A	A

In 2003, CNSC staff's evaluation of a full scale emergency exercise at Bruce A indicated that the licensee's performance met and exceeded CNSC requirements.

ENVIRONMENTAL PERFORMANCE

Site	SAFETY AREA	Grades	
		Program	Implementation
Bruce A	ENVIRONMENTAL PERFORMANCE	B	B
Bruce B	ENVIRONMENTAL PERFORMANCE	B	B

The design and implementation of programs at Bruce A and B to identify, monitor and control releases of nuclear and hazardous substances met requirements and CNSC staff's expectations.

RADIATION PROTECTION

Site	SAFETY AREA	Grades	
		Program	Implementation
Bruce A	RADIATION PROTECTION	B	B
Bruce B	RADIATION PROTECTION	B	B

The programs at Bruce to manage radiation doses to workers in 2003 met CNSC requirements and were adequately implemented.

SITE SECURITY

Site	SAFETY AREA	Grades	
		Program	Implementation
Bruce A	SITE SECURITY	B	B
Bruce B	SITE SECURITY	B	B

Programs to maintain security of the Bruce site in 2003 met the applicable requirements.

SAFEGUARDS

Site	SAFETY AREA	Grades	
		Program	Implementation
Bruce A	SAFEGUARDS	A	A
Bruce B	SAFEGUARDS	A	A

Programs at Bruce to help fulfil Canada's obligations with respect to international safeguards exceeded the applicable legal requirements and staff's expectations.

DARLINGTON

OPERATING PERFORMANCE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Darlington	OPERATING PERFORMANCE	B	B
	Organization & Plant Management	B	B
	Operations	B	B
	Occupational Health & Safety (Non-radiological)	B	B

Darlington met CNSC requirements in all three program areas. CNSC staff concluded that OPG operated Darlington safely in 2003. OPG implemented a Human Performance Improvement Plan to address issues with conformity to procedural and regulatory requirements. CNSC staff will monitor the implementation and effectiveness of these changes. Also, CNSC staff continues to follow up OPG's progress in implementing corrective actions related to significant developments in 2003:

- maintenance in the wrong unit (see Section D.7); and
- trip of Unit 2 and impairment of annulus gas system (see Section D.18).

PERFORMANCE ASSURANCE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Darlington	PERFORMANCE ASSURANCE	B	C
	Quality Management	B	C
	Human Factors	B	C
	Training	B	C

The "C" grade for implementation of the safety area reflects weakness in implementation of all three constituent programs.

Darlington experienced two implementation problems in the Quality Management program in 2003. First, the licensee self-identified the non-implementation, by site management, of the corporate program for controlling and calibrating measuring and test equipment. CNSC staff instructed OPG to provide periodic updates on the effectiveness of their corrective actions, with an expected closure by mid-2004. Second, CNSC staff cancelled a planned Type I inspection of the engineering change control program for permanent modifications. CNSC staff learned through OPG's internal assessment reports that the design portion of the program had conditions adverse to quality and that previous corrective actions had not been effective. CNSC staff felt that continuing with the inspection would not add any value when the problems were already known. Again, CNSC staff instructed OPG to provide periodic updates on the effectiveness of their corrective actions to establish confidence in the engineering change control process. The plan is to reschedule the inspection for November 2004.

The cancelled inspection of the engineering change control program was to include a review of the implementation of the process for incorporating human factors. Although Darlington has an acceptable documented process for incorporating human factors when modifications are planned and executed, CNSC staff does not have evidence that it is reliably implemented. Therefore, Darlington continues to receive a "C" for implementation of the Human Factors program. CNSC staff will examine evidence of improvement in the November 2004 inspection.

The "C" grade for Training reflects the fact that the implementation status of the SAT at Darlington does not meet staff's expectations at this time, although improvements were observed in 2003 and the licensee continues to address the issues related to exam transfer. The grade is not abnormal at this early stage of implementation.

DESIGN AND ANALYSIS

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Darlington	DESIGN AND ANALYSIS	B	B
	Safety Analysis	B	B
	Safety Issues	B	B
	Design	B	B

The following noteworthy design changes were implemented at Darlington in 2003:

- installation of new ECC injection strainers;
- conversion of the Boron tank to a Gadolinium tank for guaranteed shutdown state applications; and
- replacement of steam door cable access hatches with CamLock connectors (see Section D.6).

EQUIPMENT FITNESS FOR SERVICE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Darlington	EQUIPMENT FITNESS FOR SERVICE	B	B
	Maintenance	B	B
	Structural Integrity	B	B
	Reliability	B	B
	Equipment Qualification	B	B

There were observed improvements in the management of work under the Maintenance program at Darlington. Initiatives included improvements in parts staging and walkdowns earlier in the work management process.

Darlington was built to EQ standards but that status has not been fully maintained. The CNSC recently learned that there are some implementation issues to be resolved with respect to the June 2004 EQ deadline. Some installed components either cannot meet current EQ requirements or their qualified lives may have expired. OPG plans to replace the affected components during scheduled outages after the deadline.

EMERGENCY PREPAREDNESS

Site	SAFETY AREA	Grades	
		Program	Implementation
Darlington	EMERGENCY PREPAREDNESS	A	A

Emergency plans and programs at Darlington, as well as their implementation, exceeded CNSC requirements and expectations.

ENVIRONMENTAL PERFORMANCE

Site	SAFETY AREA	Grades	
		Program	Implementation
Darlington	ENVIRONMENTAL PERFORMANCE	B	B

The design and implementation of programs at Darlington to identify, monitor and control releases of nuclear and hazardous substances met requirements and CNSC staff expectations.

RADIATION PROTECTION

Site	SAFETY AREA	Grades	
		Program	Implementation
Darlington	RADIATION PROTECTION	A	B

The programs at Darlington to manage radiation doses to workers in 2003 exceeded CNSC staff expectations and were adequately implemented.

SITE SECURITY

Site	SAFETY AREA	Grades	
		Program	Implementation
Darlington	SITE SECURITY	B	B

Programs to maintain security of the Darlington site in 2003 met the applicable requirements.

SAFEGUARDS

Site	SAFETY AREA	Grades	
		Program	Implementation
Darlington	SAFEGUARDS	A	A

Programs at Darlington to help fulfil Canada’s obligations with respect to international safeguards exceeded the applicable legal requirements and staff’s expectations.

PICKERING A AND B

OPERATING PERFORMANCE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Pickering A	OPERATING PERFORMANCE	B	B
	Organization & Plant Management	B	B
	Operations	B	B
	Occupational Health & Safety (Non-radiological)	B	B
Pickering B	OPERATING PERFORMANCE	B	C
	Organization & Plant Management	B	C
	Operations	B	B
	Occupational Health & Safety (Non-radiological)	B	B

There were problems in 2003 at Pickering B with respect to Organization & Plant Management, where weaknesses in the implementation of programs directly contributed to very long extensions to planned outages, and indirectly contributed to an increasing number of forced outages that were mainly attributed to equipment problems in non-nuclear systems. This is reflected in the large amount of discovery work during planned outages, the large number of unplanned transients (Table 1, Section 3) and the large value of "Unplanned Capability Loss Factor" (Table 4, Section 3).

There were also *serious process failures* during 2003 at Pickering B Units 5 and 6 during the August 14 blackout; details are provided in Section D.10.1 of Appendix D. OPG management continues to align work departments and functions to attain better operating performance. CNSC staff will monitor progress on the five-year plan that OPG has established to improve the condition of station systems at Pickering B with additional maintenance and design modifications. The weakness in implementation of the Organization & Plant Management program in 2003, reflected in the "C" grade, was significant enough to assign a grade of "C" for implementation of the overall safety area of Operating Performance.

Under the Operations program, evidence gathered in the latter part of 2003 indicates that the operation of Pickering A Unit 4 has been satisfactory since its return to power in September. CNSC staff's evaluation of the Pickering B Unit 7 outage in 2003, while noting some improvements in several areas, found no definite improvement trend in the major areas of outage management (material availability, scope control, work execution process, rework, radiation and conventional safety, and contractor qualification) that have consistently and adversely affected the attainment of Pickering B's outage goals and objectives.

PERFORMANCE ASSURANCE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Pickering A	PERFORMANCE ASSURANCE	B	C
	Quality Management	B	C
	Human Factors	B	B
	Training	B	C
Pickering B	PERFORMANCE ASSURANCE	B	B
	Quality Management	B	C
	Human Factors	B	B
	Training	B	B

The “C” grade for implementation of the safety area at Pickering A reflects weaknesses in implementation in two of the three constituent programs.

The Quality Management program for Pickering A experienced problems implementing timely and effective corrective actions resulting from CNSC inspection findings and the rejection of OPG’s initial response. CNSC staff continues to monitor OPG’s progress with an expected closure by mid-2004. The Quality Management program for Pickering B showed neither improvement nor decline in 2003, the “C” grade for implementation is attributed to weaknesses that still exist in the quality assurance program for pressure boundary work.

The Human Factors program was assessed with respect to restart activities at Pickering A. OPG proposed changes to the minimum number of staff that would be present in the facility at all times; CNSC staff approved these changes. CNSC staff also reviewed the usability of selected procedures for dealing with abnormal incidents. There were no evaluations of the Human Factors program at Pickering B in 2003; therefore, the grades remain “B” for both program and implementation.

The “C” grade for Training reflects the fact that the implementation status of the SAT at Pickering A does not meet CNSC staff expectations at this time, although improvements were observed in 2003 and the licensee continues to address the issues related to exam transfer. The grade is not abnormal at this early stage of implementation.

DESIGN AND ANALYSIS

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Pickering A	DESIGN AND ANALYSIS	B	B
	Safety Analysis	B	B
	Safety Issues	B	B
	Design	B	B
Pickering B	DESIGN AND ANALYSIS	C	C
	Safety Analysis	B	B
	Safety Issues	B	B
	Design	C	C

The following noteworthy design changes were implemented at Pickering A in 2003 for the restart of Unit 4 (reference CMD 03-H7):

- shutdown system enhancement;
- seismic upgrades;
- ECC injection system strainers;
- emergency service water system improvements;
- auxiliary boiler feedwater system enhancements;
- modification to Chameleon programmable controllers in *special safety systems*; and
- replacement of digital control computers.

The following noteworthy design change was implemented at Pickering B in 2003 (reference CMD 03-H8):

- installation of ECC injection system strainers.

Pickering B is rated "C" for Design program and implementation because of numerous developments related to design in 2003, which resulted in both operational and safety problems. Two examples were the freezing of piping in the powerhouse and its adverse effects on various systems (see Section D.2), and the demineralized water leak and resulting shutdown of all four Pickering B units (see Section D.5). The August 14 blackout highlighted several design deficiencies in some systems in the plant (see Section D.10.1), such as the inadequacy of the service water system and unavailability of firewater. These issues, including their impact on Safety Analysis, will continue to be assessed in 2004. The design issues for Pickering B are significant and are the reason that both the program and implementation ratings for the overall safety area of Design and Analysis are also rated "C".

EQUIPMENT FITNESS FOR SERVICE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Pickering A	EQUIPMENT FITNESS FOR SERVICE	B	B
	Maintenance	B	B
	Structural Integrity	B	B
	Reliability	B	B
	Equipment Qualification	B	B
Pickering B	EQUIPMENT FITNESS FOR SERVICE	B	B
	Maintenance	C	C
	Structural Integrity	B	B
	Reliability	B	B
	Equipment Qualification	B	B

The number of maintenance backlogs at Pickering B did not improve in 2003, as committed by OPG; in fact, they increased. The following changes in total work orders, for the four Pickering B units, were noted between January and December 2003. Backlogs increased from:

- 3747 to 4129 for operating corrective maintenance;
- 2866 to 3244 for outage corrective maintenance;
- 0 to 20 for overdue preventive maintenance; and
- 144 to 157 for deferred preventive maintenance.

Degradation and failure of components and equipment caused or contributed to most of the plant transients and forced outages that occurred in 2003 at Pickering B. During the August 14 blackout, maintenance-related equipment problems adversely affected the performance of several systems, including the generator, boiler feedwater and service water systems. For example, due to insufficient maintenance, the emergency low pressure service water pumps could not supply the pressure and flow rate to meet the design requirements.

OPG has initiated programs that should improve the material condition of the plants when they are fully implemented. CNSC staff will continue to monitor the status of these initiatives, particularly with respect to preventive and corrective maintenance at Pickering B.

Pickering B's Structural Integrity program and implementation met staff's expectations. The following are two procedural concerns with the structural integrity implementation at Pickering B.

CNSC staff became aware that OPG had used spacers extensively during repairs and replacements to accommodate flange misalignments or changes in dimension of replacement components in service water systems. CNSC staff has expressed concerns that these components should be registered fittings, subject to all pressure boundary component requirements. The interpretation of CSA N285.0-95 does not explicitly address this issue. This issue has been brought before a CSA technical committee for interpretation, and CNSC staff will accept the outcome of the committee's deliberations.

OPG completed only 50% of the steam generator tube inspections it planned to perform on Unit 6 during 2003. Although the steam generator tubes are in relatively good condition on this unit and sufficient tubes were inspected to meet CSA code requirements, OPG should have informed CNSC staff of their intention to defer these inspections in advance.

On the topic of Reliability, all safety systems met the regulatory targets for availability, although high pressure ECC pumps and some safety support systems were unavailable for a few hours at Pickering B during the August 14 blackout.

The major part of the remaining work to meet the EQ deadline at Pickering A involves updating documentation. At Pickering B, the EQ upgrade work originally planned is on schedule, but recent findings have increased the scope of the EQ modifications on two units. OPG will complete the additional work during the next scheduled outages. Staff reviews and inspections at Pickering B related to Equipment Qualification also revealed some weaknesses in the inspection and testing of the firewater supply system.

EMERGENCY PREPAREDNESS

Site	SAFETY AREA	Grades	
		Program	Implementation
Pickering A	EMERGENCY PREPAREDNESS	A	A
Pickering B	EMERGENCY PREPAREDNESS	A	A

In 2003, CNSC staff's evaluation of a major exercise with the off-site authorities and organizations at Pickering B indicated that the licensee's performance met and exceeded CNSC requirements. Performance at Pickering A was also considered to exceed CNSC requirements.

ENVIRONMENTAL PERFORMANCE

Site	SAFETY AREA	Grades	
		Program	Implementation
Pickering A	ENVIRONMENTAL PERFORMANCE	B	B
Pickering B	ENVIRONMENTAL PERFORMANCE	B	B

At Pickering A and B, the design and implementation of programs to identify, monitor and control releases of nuclear and hazardous substances met requirements and CNSC staff expectations.

RADIATION PROTECTION

Site	SAFETY AREA	Grades	
		Program	Implementation
Pickering A	RADIATION PROTECTION	B	B
Pickering B	RADIATION PROTECTION	B	B

The programs at Pickering to manage radiation doses to workers in 2003 met CNSC requirements and were adequately implemented.

SITE SECURITY

Site	SAFETY AREA	Grades	
		Program	Implementation
Pickering A	SITE SECURITY	B	B
Pickering B	SITE SECURITY	B	B

Programs to maintain security of the Pickering site in 2003 met the applicable requirements.

SAFEGUARDS

Site	SAFETY AREA	Grades	
		Program	Implementation
Pickering A	SAFEGUARDS	A	A
Pickering B	SAFEGUARDS	A	A

Programs at Pickering to help fulfil Canada’s obligations with respect to international safeguards exceeded the applicable legal requirements and staff’s expectations.

GENTILLY-2

OPERATING PERFORMANCE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Gentilly-2	OPERATING PERFORMANCE	B	B
	Organization & Plant Management	B	B
	Operations	B	B
	Occupational Health & Safety (Non-radiological)	B	B

Some observations regarding the Operations program in 2003 were noteworthy. First, some of Hydro-Québec's approval requests were untimely. Also, the second of two planned shutdowns at Gentilly-2 in 2003 was originally planned for two months, but was extended to four months due to unplanned corrective maintenance on the turbine, as well as the greater-than-expected complexity of the repair of a cracked feeder pipe (see Section D.17). During that outage, CNSC staff conducted numerous inspection activities and observed no significant difficulties with the conduct of planned maintenance, except for the feeder repair work, which experienced several avoidable difficulties with respect to both internal and external controls.

In the area of non-radiological (conventional) safety, some weaknesses were observed in Hydro-Québec's worker protection. Radiation protection practices during the outage showed that the difficulties observed in 2002 persist, many of which relate to worker self-protection. The total internal dose due to tritium uptake was also significantly higher than expected. Despite these weaknesses, the improvement noted in the 2002 report in the areas of fire protection and housekeeping continues, even if the formalization of these practices is not complete.

PERFORMANCE ASSURANCE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Gentilly-2	PERFORMANCE ASSURANCE	C	C
	Quality Management	C	C
	Human Factors	C	C
	Training	B	B

The "C" grade for program and implementation of the safety area reflects weaknesses in two of the three constituent programs.

Under the Quality Management program, Hydro-Québec continued with the implementation of its quality system documentation in 2003, as per the initial schedule. Samples of second-level documents were verified by CNSC staff as they were completed, and modifications were requested for completeness and adherence to standards. Hydro-Québec improved the format of its quality system documents by utilizing a user-friendly and systematic approach. Follow-ups were initiated by CNSC staff related to an audit in 2000 on non-conformity, corrective action and operating experience, as well as an audit in 2001 on design modification. Document changes and answers are still required from Hydro-Québec in order to close the relevant directives and action notices. The implementation schedule for the third level of documentation—management methods, which will replace the existing documentation—was drafted. Even though the progress of the implementation of the second-level documents was slowed as a result of the extent of the annual shutdown last fall, Hydro-Québec restated its commitment to complete the overall QA implementation by the October 2004 deadline. Overall, steady improvement was observed at Gentilly-2 in both the QA program and its implementation.

The “C” grades given to Gentilly-2 for the Human Factors program are attributed to weaknesses that still exist in the design change processes. CNSC staff expects the modification process to systematically consider the needs of system users (as stated in CNSC Regulatory Guides G-276 and G-278). Gentilly-2 submitted a corrective action plan for incorporating human factors into its design change process. CNSC staff expects the modification process at Gentilly-2 to incorporate human factors by October 31, 2004.

DESIGN AND ANALYSIS

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Gentilly-2	DESIGN AND ANALYSIS	B	B
	Safety Analysis	B	B
	Safety Issues	B	B
	Design	B	B

There were no noteworthy design changes at Gentilly-2 in 2003.

EQUIPMENT FITNESS FOR SERVICE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Gentilly-2	EQUIPMENT FITNESS FOR SERVICE	B	B
	Maintenance	B	B
	Structural Integrity	B	B
	Reliability	B	B
	Equipment Qualification	B	B

A leaking feeder was detected at Gentilly-2 in 2003, but the leak rate was consistently well below the allowable value. See Section D.17 for a description of the problem, repair and follow-up investigation. Hydro-Québec is actively pursuing this issue, and CNSC staff is satisfied with its approach and progress.

Hydro-Québec will not implement a number of required EQ-related modifications until after the June 2004 deadline. Use of polyvinyl chloride (PVC)-insulated cables inside containment is still under investigation. Hydro-Québec identified the several PVC formulations used in the affected cables, performed tests on cable samples and submitted preliminary test results to CNSC.

EMERGENCY PREPAREDNESS

Site	SAFETY AREA	Grades	
		Program	Implementation
Gentilly-2	EMERGENCY PREPAREDNESS	A	A

CNSC staff evaluated a tabletop exercise at Gentilly-2 in 2003 (emergency exercise of limited scope). The Emergency Preparedness program at Gentilly-2 was judged to be stable or improving, and consequently, the rating given in the previous year was retained. A major exercise with the off-site authorities and organizations is scheduled for 2004.

ENVIRONMENTAL PERFORMANCE

Site	SAFETY AREA	Grades	
		Program	Implementation
Gentilly-2	ENVIRONMENTAL PERFORMANCE	B	B

In response to the outstanding actions from previous evaluations of the radiological environmental monitoring programs at Gentilly-2, the licensee submitted ecological risk assessment studies that satisfactorily addressed CNSC staff concerns. Further, the study results will be used by the licensee to refine the environmental monitoring program. The program at Gentilly-2, which was previously rated as below requirements, is now considered to meet CNSC requirements.

RADIATION PROTECTION

Site	SAFETY AREA	Grades	
		Program	Implementation
Gentilly-2	RADIATION PROTECTION	A	C

During 2003, Hydro-Québec continued to implement the action plan developed in 2002 in order for its employees to adhere to radiation safety procedures. Some difficulties were noted in the handling of heavy water and Hydro-Québec is dealing with the issue in a proactive manner. Work is also required in the areas of integrating radiation and industrial safety programs, self-assessment and incident trend analysis. The grade for implementation remains “C” while CNSC staff monitors the action plan for signs of improvement in these areas.

SITE SECURITY

Site	SAFETY AREA	Grades	
		Program	Implementation
Gentilly-2	SITE SECURITY	B	B

Programs to maintain security of the Gentilly-2 site in 2003 met the applicable requirements.

SAFEGUARDS

Site	SAFETY AREA	Grades	
		Program	Implementation
Gentilly-2	SAFEGUARDS	A	A

Programs at Gentilly-2 to help fulfil Canada’s obligations with respect to international safeguards exceeded the applicable legal requirements and staff’s expectations.

POINT LEPREAU

OPERATING PERFORMANCE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Point Lepreau	OPERATING PERFORMANCE	B	B
	Organization & Plant Management	B	B
	Operations	B	B
	Occupational Health & Safety (Non-radiological)	B	B

With respect to the Operations program, CNSC staff conducted a QA audit of outage work planning and management, including control of contractors on site. No serious problems in the implementation of the existing processes at Point Lepreau were identified.

PERFORMANCE ASSURANCE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Point Lepreau	PERFORMANCE ASSURANCE	C	C
	Quality Management	C	C
	Human Factors	C	C
	Training	B	C

The problems associated with Performance Assurance in both the programs and their implementation continued in 2003, particularly with respect to the Quality Management and Human Factors programs. Regarding Quality Management, NB Power continued to work on writing and implementing its QA program. Progress was slow but steady, and CNSC staff is satisfied by the process being used to develop the quality program. The NB Power QA program is clearly improving, although the grades for program and implementation for 2003 remain at "C".

Under the Human Factors program, CNSC staff continued to review activities in preparation for the refurbishment of Point Lepreau, which will include construction of a solid radioactive waste management facility. CNSC staff reviewed human factors activities that were carried out in order to ensure the design of the facility would minimize the risk of human error. In addition, CNSC staff reviewed design guides that will support modifications.

The "C" grades for Human Factors are due to weaknesses that still exist in the design change processes. CNSC staff expects the modification process of licensees to systematically consider the needs of system users (as stated in CNSC Regulatory Guides G-276 and G-278). CNSC staff will review the modification process at Point Lepreau in 2004 to confirm that deficiencies have been corrected.

The "C" grade for the Training program reflects the fact that the implementation status of the SAT does not meet staff expectations at this time, although improvements were observed in 2003 and the licensee continues to address the issues related to exam transfer. The grade is not abnormal at this early stage of implementation.

DESIGN AND ANALYSIS

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Point Lepreau	DESIGN AND ANALYSIS	B	B
	Safety Analysis	B	B
	Safety Issues	B	B
	Design	B	B

There were no noteworthy design changes at Point Lepreau in 2003.

EQUIPMENT FITNESS FOR SERVICE

Site	SAFETY AREA Program	Grades	
		Program	Implementation
Point Lepreau	EQUIPMENT FITNESS FOR SERVICE	B	B
	Maintenance	B	B
	Structural Integrity	B	B
	Reliability	B	B
	Equipment Qualification	B	B

NB Power continued to make improvements to its maintenance program through the development and implementation of an improved QA program, resulting in the issuance of key maintenance program documentation in March 2003

No leakage was detected in association with the feeder cracks found at Point Lepreau in 2003. NB Power is actively pursuing the feeder cracking issue, and CNSC staff is satisfied with its approach and progress. See Section D.13 for a description of the problem, repairs and follow-up activities.

Regarding the Equipment Qualification program, the use of cables inside containment that are insulated with PVC is under investigation. Ten samples of PVC cables were successfully tested. NB Power claims that these samples represent 92% of all PVC cables, but traceability issues still need to be resolved. NB Power is also assessing the status of the remaining 8% of the cables that are made of PVC and will subsequently decide on whether to qualify or replace them.

EMERGENCY PREPAREDNESS

Site	SAFETY AREA	Grades	
		Program	Implementation
Point Lepreau	EMERGENCY PREPAREDNESS	A	C

In 2003, CNSC staff evaluated a major exercise with the off-site authorities and organizations at Point Lepreau. The evaluation indicated that there were some weaknesses in the implementation of the Emergency Preparedness program. In particular, CNSC staff identified some weaknesses related to the habitability of response centres, training and staffing levels. NB Power has initiated actions to rectify some of the deficiencies. However, a subsequent evaluation of emergency preparedness at Point Lepreau continued to indicate some problems related to implementation. A Type I inspection of the emergency preparedness program is scheduled for 2004.

ENVIRONMENTAL PERFORMANCE

Site	SAFETY AREA	Grades	
		Program	Implementation
Point Lepreau	ENVIRONMENTAL PERFORMANCE	B	B

Outstanding actions related to environmental monitoring at Point Lepreau were satisfactorily resolved by the licensee in 2003. The program at Point Lepreau, which was previously rated as below requirements, is now considered to meet the CNSC requirements.

RADIATION PROTECTION

Site	SAFETY AREA	Grades	
		Program	Implementation
Point Lepreau	RADIATION PROTECTION	B	B

The programs at Point Lepreau to manage radiation doses to workers in 2003 met CNSC requirements and were adequately implemented.

SITE SECURITY

Site	SAFETY AREA	Grades	
		Program	Implementation
Point Lepreau	SITE SECURITY	B	B

Programs to maintain security of the Point Lepreau site in 2003 met the applicable requirements.

SAFEGUARDS

Site	SAFETY AREA	Grades	
		Program	Implementation
Point Lepreau	SAFEGUARDS	A	B

Electrical power to IAEA-installed safeguards surveillance equipment was temporarily interrupted at Point Lepreau, resulting in the loss of surveillance data. Because of the internal facility communication protocol, the safeguards officer was not notified of the power interruption for six days. The power was restored within four hours of discovering that the safeguards equipment was affected. The IAEA had to re-verify the nuclear material in the spent fuel storage bays as a result of the power interruption.

SECTION 3

TABLES OF PERFORMANCE INDICATORS

The performance indicators (PIs) tabulated in this section are defined in detail in Regulatory Standard S-99, Reporting Requirements for Operating Nuclear Power Plants.

The purpose of the "Number of Unplanned Transients" PI is to indicate the number of reactor power transients due to equipment failures or operator errors while the reactor is not in a guaranteed shutdown state.

The "Number of Unplanned Transients" PI is illustrated in Tables 1, 2 and 3. This PI shows the number of manual or automatic power reductions from actuation of the shutdown, *stepback* or *setback* systems (note that Pickering A does not have a stepback system). The PI also includes the number of hours in *guaranteed shutdown state* (GSS) for the reactors. Unexpected power reductions may be indicative of problems within the plant and may place unnecessary strain on systems. Many of the unplanned transients in 2003 were setbacks, which typically pose little risk to plant operations. The significant transients are described in the significant development reports (see Appendix D).

Table 1: Number of Unplanned Transients for 2003

Station	GSS Hrs	Unplanned Transients at Sites in 2003			
		Trips	Stepbacks	Setbacks	Total
Bruce A	0 *	0	1	0	1
Bruce B	4,306	1	7	0	8
Darlington	4,906	4	5	1	10
Pickering A	29,968	5	NA	2	7
Pickering B	5,241	8	0	6	14
Gentilly-2	2,831	0	0	2	2
Point Lepreau	669	1	0	0	1
Total for Industry	47,922	19	13	11	43

* only reported for Unit 4 in 4th quarter of 2003

Tables 2 and 3 show the trends of this PI for the industry since 2000. The number of unplanned transients was greater in 2003 compared to previous years. This was partly due to restart activities at Pickering A (where none of the transients in the first three quarters of 2003 occurred at full power) and transients resulting from the August 14 blackout. The Canadian industry average over the last four years (excluding Bruce A and Pickering A) has been one trip or stepback for approximately every 7,100 hours of criticality. The international performance target is one reactor trip per 7,000 hours of reactor operation.

Table 2: Trend Details of Number of Unplanned Transients for Industry

Year	GSS Hrs	Unplanned Transients in Industry			
		Trips	Stepbacks	Setbacks	Total
2000	57,788	5	4	2	11
2001	41,341	8	5	10	21
2002	51,503	6	1	13	17
2003	47,922	19	13	11	43

Table 3: Trends of Number of Unplanned Transients for Stations

Station	Unplanned Transients			
	2000	2001	2002	2003
Bruce A	NA	NA	NA	1
Bruce B	5	3	6	8
Darlington	1	5	1	10
Pickering A	0	0	0	7
Pickering B	3	12	6	14
Gentilly-2	1	0	2	2
Point Lepreau	1	1	2	1
Total for Industry	11	21	17	43

The purpose of the "Unplanned Capability Loss Factor" PI is to indicate how a unit is managed, operated, maintained in order to avoid unplanned outages.

Tables 4 and 5 show the "Unplanned Capability Loss Factor" PI, which is the percentage of the reference electrical output for the station that was lost during the period due to unplanned circumstances. In addition to being an economic indicator, it is a reflection of overall management of the plant. This factor has generally been higher for Pickering B than any other station since 2000, and especially in 2003 (see Table 5). The August 14 blackout was responsible for some of the lost output in the third quarter for sites in Ontario.

Table 4: Unplanned Capability Loss Factor for 2003

Station	Unplanned Capability Loss Factor (%)				
	Quarter				For Year
	Q1	Q2	Q3	Q4	
Bruce B	0.0	0.0	2.1	13.1	3.8
Pickering B	20.5	22.5	27.4	6.2	19.1
Darlington	2.9	2.0	3.9	8.3	4.3
Gentilly-2	0.5	0.2	0.0	0.0	0.2
Point Lepreau	2.4	0.0	0.0	13.1	3.9

Table 5: Trend Details of Unplanned Capability Loss Factor for Industry

Station	Unplanned Capability Loss Factor (%)			
	Year			
	2000	2001	2002	2003
Bruce B	3.8	1.3	6.4	3.8
Pickering B	15.4	9.6	7.2	19.1
Darlington	7.8	5.6	4.9	4.3
Gentilly-2	0.0	0.0	0.0	0.2
Point Lepreau	0.0	14.3	9.2	3.9

The purpose of the "Non-Compliance Index" PI is to indicate the number of occurrences where the operation of the station failed to comply with its licence conditions, or with the *Nuclear Safety and Control Act* and Regulations.

Tables 6, 7 and 8 illustrate the "Non-Compliance Index" PI. Non-compliances are categorized by type as follows:

- a = number of non-compliances with the operating policies and principles that are referenced in the licence;
- b = number of non-compliances with the radiation protection requirements that are referenced in the licence;
- c = number of non-compliances with the minimum shift complement that are referenced in the licence;
- d = number of other non-compliances with the licence; and
- e = number of non-compliances with the NSCA and Regulations.

All non-compliances are evaluated by CNSC staff. Table 6 shows that most non-compliances in 2003 fell in the categories of "b" and "d". Pickering had the most non-compliances in 2003 (Table 6), as well as in previous years (Table 8). This is partly explained by the size of the operation (four reactors in rehabilitation and four operating at the Pickering site). It should also be noted that the non-compliances are relative to the different requirements at each site, including different operating policies and principles, radiation requirements, designs, licence conditions, practices, etc.

Table 6: Non-Compliance Index for 2003

Station	Non-Compliances by Type					
	a	b	c	d	e	Total
Bruce A	9	39	1	69	2	120
Bruce B	2	36	6	34	1	79
Pickering A & B	102	75	3	74	28	282
Darlington	17	31	0	14	8	70
Gentilly-2	4	4	0	3	2	13
Point Lepreau	8	1	0	9	9	27

Tables 7 and 8 show that the number of non-compliances in the industry in 2003 was consistent with the previous three years.

Table 7: Trend Details of Non-Compliance Index for Industry

Year	Non-Compliances by Type					
	a	b	c	d	e	Total
2000	307	109	31	115	50	612
2001	239	161	3	169	17	589
2002	219	140	13	222	24	618
2003	142	186	10	203	50	591

Table 8: Trends of Non-Compliance Index for Stations

Station	Total Non-Compliances			
	2000	2001	2002	2003
Bruce A	42	9	24	120
Bruce B	219	123	124	79
Pickering A & B	238	295	337	282
Darlington	63	110	58	70
Gentilly-2	22	18	20	13
Point Lepreau	28	34	55	27
Total for Industry	612	589	618	591

The purpose of the “Accident Severity Rate” PI is to monitor performance in meeting nuclear industry standards in the area of worker safety.

Tables 9, 10 and 11 illustrate the “Accident Severity Rate” PI, which measures the total number of days lost to injury for every 200,000 person hours worked at the site. The rates for all the licensees in 2003 were comparable to previous years (Table 11). The severity rate tends to be low, particularly for Darlington and Point Lepreau. However, it remains greater for Gentilly-2. Caution is advised when comparing licensees due to the differences among organizations in the definitions of industrial accidents, jurisdiction of worker safety, interpretations of lost time associated with chronic health problems, etc. For example, depending on work agreements, a worker with a similar back injury might be assigned to non physical work by one licensee, while classified as a lost time accident by another licensee. At Gentilly-2, none of the lost-time accidents in 2003 were attributed to the negligence of Hydro-Québec. Some of the lost-time accidents, for instance, were attributed to back pain rather than particular mishaps. Regarding the “Accident Severity Rate” PI in Table 9 (relevant to Occupational Health & Safety), Pickering B exceeded one of its accident targets due to an off-site vehicle accident (an employee on training) that led to an extended absence.

Table 9: Accident Severity Rate for 2003

Site	Days Lost	Person Hours	Accident Severity
Bruce A & B	134	6,362,845	4.2
Pickering A & B	90	4,809,538	3.7
Darlington	8	2,722,215	0.6
Gentilly-2	139	1,361,175	20.4
Point Lepreau	1	1,357,111	0.1

Table 10: Trend Details of Accident Severity Rate for Industry

Year	Days Lost	Person Hours	Accident Severity
2000	462	19,186,826	4.8
2001	468	19,514,814	4.8
2002	350	17,579,865	4.0
2003	372	16,612,884	4.48

Table 11: Trends of Accident Severity Rate for Stations

Site	Accident Severity Rate			
	2000	2001	2002	2003
Bruce A & B	3.8	9.7	4.8	4.2
Pickering A & B	3.9	0.7	1.4	3.7
Darlington	8.0	0.7	0.0	0.6
Gentilly-2	6.5	18.0	25.2	20.4
Point Lepreau	1.3	8.5	0.0	0.1

The purpose of the "Number of Pressure Boundary Degradations" PI is to indicate the number of pressure boundary degradations which have occurred at the station and to monitor the performance in meeting nuclear industry codes and standards.

Tables 12, 13 and 14 illustrate the "Number of Pressure Boundary Degradations" PI. Degradations are defined as instances where limits in relevant design or inspection criteria are exceeded. The "class" that is referred to is the code classification of nuclear systems, whereas "conventional" refers to non-nuclear systems. The number of degradations in 2003 was consistent with the last few years (Table 13). The number of degradations at Bruce A in 2003, compared with previous years (Table 14), is at least partly related to the increased activity involved in restarting Units 3 and 4.

Table 12: Pressure Boundary Degradations for 2003

Station	# Pressure Boundary Degradations by Type					
	class 1	class 2	class 3	class 4	conventional	Total
Bruce A	24	8	12	0	87	131
Bruce B	9	0	7	1	92	109
Darlington	0	0	0	0	59	59
Pickering A & B	1	2	8	0	89	100
Gentilly-2	0	0	0	0	0	0
Point Lepreau	3	0	1	0	6	10

Table 13: Trend Details of Pressure Boundary Degradations for Industry

Year	# Pressure Boundary Degradations by Type					
	class 1	class 2	class 3	class 4	conventional	Total
2000	54	8	51	2	379	494
2001	24	9	30	1	281	345
2002	18	11	37	0	261	327
2003	37	10	28	1	333	409

Table 14: Trends of Pressure Boundary Degradations for Stations

Station	Total # Pressure Boundary Degradations			
	2000	2001	2002	2003
Bruce A	51	21	18	131
Bruce B	197	47	71	109
Darlington	65	80	91	59
Pickering A & B	125	155	109	100
Gentilly-2	11	3	3	0
Point Lepreau	45	39	35	10

The purpose of the “Number of Missed Mandatory Safety System Tests” PI is to indicate successful completion of tests required by licence condition, including those referenced in documents submitted in support of a licence application, i.e. to monitor performance in meeting regulatory and licensee availability requirements.

Tables 15, 16 and 17 show the “Number of Missed Mandatory Safety System Tests” PI. This PI represents the ability of licensees to successfully complete routine tests on systems related to safety. Approximately 70,000 of these tests were performed throughout the industry in 2003. The extremely small number of missed tests indicates a consistent industry commitment to test its safety systems on a regular basis. CNSC staff reviewed each missed test in 2003 and found that none significantly impacted on safety.

Table 15: Missed Mandatory Safety System Tests for 2003

Station	Total # Tests	Missed Mandatory Safety System Tests			
		Special	Standby	Safety Related	Total
Bruce B	31,764	0	0	0	0
Darlington	10,800	0	0	0	0
Pickering A	8,186	0	0	0	0
Pickering B	8,238	1	2	2	5
Gentilly-2	No data	1	0	1	2
Point Lepreau	5,315	0	0	0	0
Total	64,303	2	2	3	7

Table 16: Trend Details of Missed Mandatory Safety System Tests for Industry

Year	Total # Tests *	Total # Missed Mandatory Safety System Tests			
		Special	Standby	Safety Related	Total
2000	no data	11	6	25	42
2001	52,841	2	0	4	6
2002	63,864	3	1	0	4
2003	64,303	2	2	3	7

* Total # Tests excludes number for Gentilly-2

Table 17: Trend of Missed Mandatory Safety System Tests for Stations

Station	Missed Mandatory Safety System Tests			
	2000	2001	2002	2003
Bruce B	1	0	0	0
Darlington	32	4	0	0
Pickering A	0	0	0	0
Pickering B	6	2	1	5
Gentilly-2	0	0	1	2
Point Lepreau	3	0	2	0
Total for Industry	42	6	4	7

The purpose of the “Radiation Occurrence Index” PI is to indicate the number and weighted severity of radiation occurrences which have taken place at the station, thus monitoring the performance in meeting the CNSC’s expectations in the area of worker radiation protection.

Tables 18, 19 and 20 show the “Radiation Occurrence Index” PI. The index and its components are defined and calculated as follows:

a = number of occurrences, after decontamination attempts, of fixed body contamination > 50 kBq/m^2

b = number of occurrences of unplanned acute whole body doses from external exposure > 5 mSv

c = number of occurrences of intake of radioactive material with effective dose > 2 mSv (normalized to 2 mSv)
d = number of occurrences of acute or committed dose in excess of specified limits

Radiation Occurrence Index = a + 5b + 5c + 50d

The "weight" of each component in the formula indicates the relative significance of the various types of occurrence. The value of the index has generally been steady for the industry since 2000 (Table 19), with no doses in excess of specified limits (value of "d"). For some licensees, no occurrences of any type occurred (Table 20).

Table 18: Radiation Occurrence Index for 2003

Station	Radiation Occurrence				
	a	b	c	d	Index
Bruce A	0	0	0	0	0
Bruce B	0	0	0	0	0
Darlington	0	0	0	0	0
Pickering A & B	0	0	0	0	0
Gentilly-2	2	0	6.7	0	35.5
Point Lepreau	0	0	0	0	0

Table 19: Trend Details of Radiation Occurrence Index for Industry

Year	Radiation Occurrence (Totals)				
	a	b	c	d	Index
2000	0	0	9.5	0	47.4
2001	1	0	8.8	0	45.2
2002	0	0	4.4	0	22.0
2003	2	0	6.7	0	35.5

Table 20: Trend of Radiation Occurrence Index for Stations

Station	Radiation Occurrence Index			
	2000	2001	2002	2003
Bruce A	0	0	0	0
Bruce B	0	17.2	13.2	0
Darlington	0	0	0	0
Pickering A & B	12.4	0	8.8	0
Gentilly-2	22.2	27.0	0	35.5
Point Lepreau	12.8	1.0	0	0

APPENDIX A
GLOSSARY OF TERMS

These terms are italicized when first used in the text.

Action Item
A numbered tracking system used by CNSC staff to control issues requiring licensee attention.

Commission
A corporate body of not more than seven members, established under the NSCA and appointed by Governor in Council, to:

- regulate the development, production and use of nuclear energy, the production, possession, use and transport of nuclear substances;
- regulate the production, possession and use of prescribed equipment and prescribed information;
- implement measures respecting international control of the development, production, transport and use of nuclear energy and nuclear substances, including those respecting the non-proliferation of nuclear weapons and nuclear explosive devices; and
- disseminate scientific, technical and regulatory information concerning the activities of the CNSC and the effects on the environment and on the health and safety of persons, of the development, production, possession, transport and uses referred to above.

Commission Member Documents (CMDs)
Documents prepared for Commission hearings and meetings by CNSC staff, proponents and intervenors. Each CMD is assigned a specific identification number.

Derived Release Limit
A limit imposed by the CNSC on the release of a radioactive substance from a licensed nuclear facility such that compliance with the derived release limit gives reasonable assurance that the regulatory dose limit is not exceeded.

Environmental qualification (EQ)
A program that establishes an integrated and comprehensive set of requirements that provide assurance that essential equipment can perform as required if exposed to harsh conditions, and that this capability is maintained over the life of the plant.

Feeder
There are several hundred channels in the reactor that contain fuel. The feeders are pipes attached to each end of the channels used to circulate heavy water coolant from the fuel channels to the steam generators.

Guaranteed shutdown state (GSS)
A method for ensuring that the reactor is shut down. It includes adding a substance to the reactor moderator which absorbs neutrons and removes them from the fission chain reaction, or draining the moderator from the reactor.

International Atomic Energy Agency (IAEA)
A United Nations' agency, it establishes a system of safeguards to ensure that member states do not divert nuclear materials to non-peaceful activities. It also provides an international forum for nuclear safety.

Lay-up state
A special configuration into which a plant is placed to prevent system and component degradation during extended periods of shutdown.

Pressure tubes
Tubes that pass through the calandria and contain 12 or 13 fuel bundles. Pressurized heavy water flows through the tubes, cooling the fuel.

Root-cause analysis
An objective, structured, systematic and comprehensive analysis that is designed to determine the underlying reason(s) for a situation or event, and that is conducted with a level of effort that is consistent with the safety significance of the event.

Safeguards
An international program of monitoring and inspection carried out by staff of the International Atomic Energy Agency. Safeguards ensure that nuclear materials at the plant are not diverted to non-peaceful uses.

Serious process failures
A failure of a process system, component or structure:
(a) that leads to a systematic fuel failure or a significant release from the nuclear power plant, or
(b) that could lead to a systematic fuel failure or a significant release in the absence of action by any special safety system.

Setback
A system designed to automatically reduce reactor power at a slow rate if a problem occurs. The setback system is part of the reactor-regulating system.

Special safety system
The shutdown system no.1, the shutdown system no. 2, the containment system, or the emergency core cooling system, of a nuclear power plant.

Steam generator
A heat exchanger that transfers heat from the heavy water coolant to ordinary water. The ordinary water boils, producing steam to drive the turbine. The steam generator tubes separate the reactor coolant from the rest of the power-generating system.

Stepback
A system designed to automatically reduce reactor power at a fast rate if a problem occurs. The stepback system is part of the reactor-regulating system.

Systematic approach to training (SAT)
A logical progression from the identification of training needs and competencies required to perform a job, to the development and implementation of training to achieve these competencies and to the subsequent evaluation of this training.

APPENDIX B
ACRONYMS

These acronyms are also defined when first used in the text.

AECL	Atomic Energy of Canada Limited
CT	calandria tube
CMD	Commission Member Document
CNSC	Canadian Nuclear Safety Commission
CSA	Canadian Standards Association
DBA	design basis accident
DDT	deflagration-to-detonation transition
ECC	emergency core coolant
EFADS	emergency filtered air discharge system
EQ	environmental qualification
FAC	flow accelerated corrosion
GAI	generic action item
GSS	guaranteed shutdown state
HTS	heat transport system
IAEA	International Atomic Energy Agency
IST	industry standard toolset
LAC	local air cooler
LOCA	loss of coolant accident
LOECC	loss of emergency core coolant
LLOCA	large loss of coolant accident
NB Power	New Brunswick Power
NSCA	Nuclear Safety and Control Act
OPG	Ontario Power Generation
PAR	passive autocatalytic recombiner
PEV	powerhouse emergency venting
PHT	primary heat transport
PI	performance indicator
PT	pressure tube
PVC	polyvinyl chloride
QA	quality assurance
SAM	severe accident management
SAT	Systematic approach to training
SCC	stress corrosion cracking
SDR	significant development report
SDS	shutdown system
SSC	structures, systems and components
SUI	start-up instrumentation
USNRC	United States Nuclear Regulatory Commission

APPENDIX C
RATING SYSTEM

Grades are assigned for both design of the program and its implementation/performance for each safety area and for programs within the safety area

A - Exceeds requirements Assessment topics or programs meet and consistently exceed applicable CNSC requirements and performance expectations. Performance is stable or improving. Any problems or issues that arise are promptly addressed, such that they do not pose an unreasonable risk to the maintenance of health, safety, security, environmental protection, or conformance with international obligations to which Canada has agreed.
B - Meets requirements Assessment topics or programs meet the intent or objectives of CNSC requirements and performance expectations. There is only minor deviation from requirements or the expectations for the design and/or execution of the programs, but these deviations do not represent an unreasonable risk to the maintenance of health, safety, security, environmental protection, or conformance with international obligations to which Canada has agreed. That is, there is some slippage with respect to the requirements and expectations for program design and execution. However those issues are considered to pose a low risk to the achievement of regulatory performance requirements and expectations of the CNSC.
C – Below requirements Performance deteriorates and falls below expectations, or assessment topics or programs deviate from the intent or objectives of CNSC requirements, to the extent that there is a moderate risk that the programs will ultimately fail to achieve expectations for the maintenance of health, safety, security, environmental protection, or conformance with international obligations to which Canada has agreed. Although the risk of failing to meet regulatory requirements in the short term remains low, improvements in performance or programs are required to address identified weaknesses. The licensee or applicant has taken, or is taking appropriate action.
D – Significantly below requirements Assessment topics or programs are significantly below requirements, or there is evidence of continued poor performance, to the extent that whole programs are undermined. This area is compromised. Without corrective action, there is a high probability that the deficiencies will lead to an unreasonable risk to the maintenance of health, safety, security, environmental protection, or conformance with international obligations to which Canada has agreed. Issues are not being addressed effectively by the licensee or applicant. The licensee or applicant has neither taken appropriate compensating measures nor provided an alternative plan of action.
E – Unacceptable Evidence of either an absence, total inadequacy, breakdown, or loss of control of an assessment topic or a program. There is a very high probability of an unreasonable risk to the maintenance of health, safety, security, environmental protection, or conformance with international obligations to which Canada has agreed. An appropriate regulatory response, such as an order or restrictive licensing action has been or is being implemented to rectify the situation.

APPENDIX D
SIGNIFICANT DEVELOPMENTS INVOLVING POWER REACTORS IN 2003

All dates are for 2003 unless otherwise noted.

D.1 Neutron Detectors at Bruce B Unit 7

D.1.1 Description of Development (Reference CMD 03-M14)

During the approach to criticality of Unit 7 in December following the 2002 planned outage, the start-up instrumentation (SUI) did not respond to the removal of the moderator poison. At very low powers, the SUI gives an indication of the rate of change of neutron flux which increases as poison is removed. In this instance, the SUI neutron count rate did not increase as expected, but remained constant. Bruce Power returned the unit to over-poisoned guaranteed shutdown state (GSS).

The preliminary findings of the investigation by Bruce Power revealed that the SUI had, in fact, not given a true indication of neutron flux from November 30 until December 23, 2002. This was a non-compliance with the Bruce B operating policies and principles which require a continuous indication of level and rate of change of neutron flux. The SUI is connected to shutdown system 1 (SDS 1) and provides a backup trip, which was also impaired, representing a reduction of defense-in-depth. The credited SDS 1 and 2 trips were operational and, based on tests early in the outage and later in the startup, were not impaired.

The preliminary findings of the Bruce Power investigation pointed to an inadequate understanding by Bruce B staff of the SUI neutron detectors (Helium-3). They were incorrectly installed at an inappropriate time and incorrectly calibrated. In addition, training on the use of SUI appeared to be inadequate. The operating documentation, detector installation procedures, and system testing procedures were also found to be inadequate.

D.1.2 Follow-up

CNSC staff verified that all causes of this serious event were identified by Bruce Power. Bruce Power took effective corrective action to prevent a repetition of this event and the action item was closed.

D.2 Setback of Unit 6 at Pickering B and Complications due to Cold Temperatures in Powerhouse Following Emergency Vent Opening

D.2.1 Description of Development (Reference CMD 03-M14)

On January 26, Unit 6 at Pickering B set-back and subsequently poisoned-out as a result of a condenser cooling water pump trip. On January 27, the turbine generator in Unit 5 was manually tripped by the operators in response to an apparent failure of a generator seal, which can result in a release of hydrogen gas into the plant. To address the potential fire hazard, operating staff followed procedures and opened the powerhouse emergency venting (PEV) panels to vent any hydrogen from the turbine building. The extremely cold outside air (below -20ºC) rapidly entered the powerhouse via the PEV panels and reduced the temperature inside the building, particularly on the north side of the powerhouse where the PEV panels are located. The cold air impaired the PEV panel logic and relays, and difficulties were encountered in closing the panels. Eventually the PEV panels were manually blocked.

The low temperature in the powerhouse also affected a number of other systems. As a result of freezing of water in small bore pipes and instrument lines, the deaerator level control, fire protection systems, auxiliary condensate extraction pumps and turbine exhaust trip logic were adversely affected. Difficulties not associated with the event were also encountered in bringing the auxiliary heating boiler into service.

The affected systems have since all been repaired and returned to service. CNSC staff was generally satisfied with Ontario Power Generation's (OPG) response to these events, but followed up with a more in-depth assessment of the event.

D.2.2 Follow-up (Reference CMD 03-M56)

CNSC staff continued to assess the results of OPG's investigation and corrective action plan for the long term. Staff questioned OPG about safety issues related to the event, including the adequacy of surveillance and maintenance of the PEV, OPG's assessment of the impairment level of the PEV system during the event and guidance for removing a safety support system from service. CNSC staff has completed the review of OPG's response and is satisfied with OPG's corrective action plan.

D.2.3 Additional Follow-up

CNSC staff also considers that OPG has adequately addressed the initial malfunction in this event, which was contact misalignment in the control system for the generator seal oil. The preventative maintenance program was modified to include this type of relay, and the relays were repaired and maintained on all units. OPG is also considering implementation of a design change to minimize the occurrence of generator trips due to single component failures.

D.3 Emergency Core Coolant System Check Valves at Bruce A and B

D.3.1 Description of Development and Initial Follow-up (Reference CMD 03-M22)

On February 28, Bruce Power found problems with the testing mechanisms on some check valves in the Bruce A emergency core coolant (ECC) system that could have limited their ability to function adequately. The check valves would not fully open in one direction, reducing the ECC flow when the system was operating in recovery mode. There was no impact on the safe operation of Bruce A since all units were shut down.

Bruce Power employees working at Bruce A promptly informed Bruce B of the situation. Bruce Power identified that the ECC system at Bruce B uses six check valves of a similar type, so a series of tests on the Bruce B ECC system check valves was performed to verify their condition. The tests revealed that three out of the six check valves were fully operational and that the deficiency observed at Bruce A was present in the remaining three check valves. Bruce Power promptly implemented interim compensatory measures and performed additional tests to further confirm the actual condition of the check valves.

Bruce Power promptly informed CNSC staff of the discovery of this condition. CNSC staff immediately assessed the situation to determine if Bruce Power's actions were adequate and to decide if additional action was required. CNSC staff concluded that the Bruce B ECC system continued to be effective, but with a slight reduction in reliability. CNSC staff considered that the operation of the Bruce B units continued to be safe and that the decisions and actions taken by Bruce Power were appropriate.

CNSC staff followed up with power reactor licensees at other sites to confirm that they were aware of this situation and that they were also taking compensatory measures as necessary.

D.3.2 Additional Follow-up

The compensatory action—repair of the Bruce A check valves—was completed prior to the restart of Units 3 and 4. The corrective actions were completed at Bruce A, and will be completed at Bruce B during the vacuum building outage in fall 2004.

D.4 Pickering B Unit 8 Trip Due to Fuse Removal Error

D.4.1 Description of Development (Reference CMD 03-M22)

On March 11, two maintainers entered a panel of Unit 8 to replace a fuse for a CO_2 monitor (8-5424-Bus 2C Fuse 28). The maintainers were unaware that the panel also contained a Bus 1C fuse and pulled 8-5424-Bus 1C Fuse 28 by mistake. This happened despite the fact that the first maintainer stated, as required by the procedure, that he was pulling Fuse 2C and the other maintainer confirmed that it was the correct fuse. The pulling of the wrong fuse resulted in a temporary loss of boiler level control to one quadrant of boilers, which, in turn, resulted in SDS 2 shutting down the reactor as per plant design.

OPG had previously made plans to inspect the Unit 8 turbine blades in the event of an unplanned outage; these additional inspections delayed the return to service of the unit.

D.4.2 Follow-up

OPG modified the maintenance procedure and also made a number of improvements to assist maintenance staff with correct component identification. CNSC staff considers these corrective actions appropriate for this event.

D.5 Unplanned Shutdown of Pickering B Units Following Demineralized Water Leak

D.5.1 Description of Development (Reference CMD 03-M22)

Demineralized water is used at Pickering B to provide make-up water to steam and condensate systems to replace water lost as a result of steam generator blow down, leakage from the systems, and diversion of hot water from the feed and condensate systems during winter months to control ice in the station intake.

On March 12, OPG staff detected a leak in the demineralized water header at Unit 5. There were no safety or environmental concerns associated with the leakage, but the location of the crack was such that it could not be isolated for repair without stopping the flow of demineralized water to all the Pickering units. Therefore, all four units at Pickering B were shut down to carry out the repair. At the time of the discovery of the crack, Unit 7 was already shut down for planned maintenance and Unit 8 was still shut down as a result of that reactor having tripped the previous day. Units 6 and 5, which were operating at full power at the time the crack was discovered, were shut down in an orderly fashion prior to implementing the necessary repairs.

D.5.2 Follow-up

OPG's analysis of this event resulted in an addition to the preventive maintenance program, correction of operating procedures and a proposal for a design modification. CNSC staff is satisfied with these corrective actions.

D.6 Darlington Steam Door Modifications

D.6.1 Update (Reference CMD 03-M28)

At the public hearings for the renewal of the Darlington operating licence in early 2003, CNSC staff reported that OPG had installed small hatches in some steam doors to facilitate the temporary passage of cabling into the rooms. CNSC staff found this to be unacceptable and instructed OPG to return the steam doors to the original design function and submit a proposal that would allow testing and maintenance of equipment in steam protected rooms without leaving the door open or unlatched. In its decision on the licence renewal, the Commission asked that CNSC staff keep it informed on progress of the issue of the steam door hatches.

On March 21, OPG submitted a request to CNSC staff for approval to modify 23 steam doors. The modifications involved removal of the hatches and installation of CamLock power connectors between batteries inside the steam-protected rooms and loads outside the rooms.

Six other steam doors with hatches were to be replaced with doors of the original design later in 2003.

D.6.2 Additional Follow-up

The modifications were completed and all doors were inspected by CNSC staff.

D.7 Maintenance in Wrong Unit at Darlington (Reference CMD 03-M28)

On April 16, maintenance personnel started to remove bolt fasteners on a man-way access cover on the condensate storage tank in Unit 3. However, the work was assigned for Unit 4. The condensate storage tank has a capacity of 200,000 litres of water, is vented to the atmosphere and has a head of water at the man-way cover of approximately 2.5 metres. As the bolts were loosened, water leaked from the man-way cover joint, so the maintainer re-tightened the bolts as per standard practice. At approximately the same time, an operator briefed to look for such occurrences on the non-outage units questioned the maintainer about working on the wrong unit.

The work was stopped and the operator reported the situation to the Unit 3 authorized nuclear operator in the main control room. No injuries occurred and there were no other safety consequences of this event.

OPG staff graded the event as Significance Level 3—"an event or adverse condition, which is not significant by itself but which has the potential to be more significant or which may be the precursor to a more significant event". OPG also determined that the event deserved a Resolution Category B—"requiring an individual root cause analysis to determine corrective action plan".

A number of actions were identified to prevent similar events:

- use of better three-way communication;
- pre-job briefings that highlight the correct unit to be worked on;
- placing of physical barriers between an outage unit and running units;
- restricting outage maintenance staff to using only the outage unit elevator; and
- ensuring that operations staff identifies equipment to be worked on by clearly tagging it and ensuring that maintenance staff works only on tagged devices.

CNSC staff met with OPG regulatory affairs personnel to ascertain whether the event was reportable or not (through CNSC Standard S-99), and to discuss any safety culture concerns that this event, together with similar events in the past, may have raised. CNSC staff has updated its compliance database for potential input to planned program evaluations. Finally, since this incident, OPG has improved the identification of units.

D.8 Pickering B Unit 5 Trip due to Controlling Computer Fault

D.8.1 Description of Development (Reference CMD 03-M28)

Unit 5 was safely shut down on May 1 following a reactor trip caused by a fault on the controlling computer. Due to the nature of the fault, the transfer of control to the standby computer was delayed, resulting in a SDS 1 trip on high primary heat transport (PHT) pressure.

D.8.2 Follow-up

OPG performed an engineering analysis to identify software changes to improve the ability of the computers to respond to hardware failures to prevent unit transients. Based on these results, appropriate and feasible modifications will be implemented. OPG has also decided to reinstate a project to establish the design basis of the computer control programs at Pickering A and B. CNSC staff considers these actions to be comprehensive and appropriate and will continue to monitor their implementation.

D.9 Blockage of Gentilly-2 Condenser Intake Filters

D.9.1 Description of Development (Reference CMD 03-M28)

On April 16, the station experienced a series of manual reductions to 59% power as operators took action to regain full capacity of the condenser following an algae run in the St-Lawrence

River. The algae blocked filters on the cooling water intake to the condenser, threatening the availability of the reactor's primary heat sink. Ensuing clean-up operations enabled the reactor to return to full power the next day.

D.9.2 Follow-up

Hydro-Québec provided responses to CNSC staff's comments on its significant event report and staff is currently reviewing them.

D.10 Effect of August 14 Blackout on Power Reactors (Reference CMD 03-M56)

D.10.1 Pickering A and B

The loss of the electrical grid caused the turbine generators on Units 5 and 6 to trip, which caused the reactors in Units 5 and 6 to trip on both SDS 1 and SDS 2. The Unit 8 reactor automatically set back and was being stabilized at 20% power when the unit further set back to 2% power. It subsequently tripped on SDS 1, caused by low boiler feedline pressure due to a power mismatch between the reactor and the turbine. Unit 7 was at 0.09% power at the time of the blackout and was manually tripped in accordance with procedures.

Unit 4 at Pickering A was at low power and preparing to synchronize to the grid for the first time since its restart when the blackout occurred. The reactor automatically tripped on low flow and low pressure of the PHT system. Units 1, 2 and 3 were in lay-up state.

The blackout resulted in a serious process failure in Units 5 and 6 when forced circulation of the PHT system was lost at high power. Unit 8 was able to reduce power to below 2% before it lost forced circulation. The high pressure ECC system at both Pickering A and B was unavailable for 5.5 hours because of loss of power to supply the high pressure pumps. In addition, emergency high pressure service water restoration for all Pickering B units was delayed because of low suction pressure for the emergency high pressure service water pumps. Manual operator intervention was required to restore some pumps back to service. The standby generators started automatically and supplied the required Class III loads. All units at Pickering were cooled and de-pressurized within 12 hours and then placed in GSS.

D.10.2 Follow-up for Pickering B (Reference CMD 04-M4)

CNSC staff reported to the International Atomic Energy Agency that the August 14 blackout was a Level 2 incident on the International Nuclear Event Scale. This is an incident with a significant failure in safety provisions but with sufficient defence-in-depth remaining to cope with additional failures.

D.10.3 Additional Follow-up for Pickering B

CNSC staff has conducted an independent investigation to determine whether OPG was adequately addressing all issues resulting from the assessment of Pickering B systems and equipment performance during the blackout. Some additional design and equipment condition issues were identified by this investigation and OPG has been asked to correct them. CNSC staff will continue to closely monitor OPG's corrective actions in response to both the OPG and CNSC investigations of this event.

D.10.4 Darlington

Following the load rejection resulting from loss of the electrical grid, Units 1 and 2 automatically reduced power. The control room operators completed the required system reviews and determined it was safe to place the units in poison prevent mode, which allows the units to be returned to power quickly, if needed. However, the shift manager was not able to carry out the required review in the time available, so the units were manually shut down.

Unit 3 automatically reduced power on load rejection. The control room shift supervisors were able to complete their independent system checks, and the reactor was placed in poison prevent mode. Unit 3 was later reconnected to the grid, supplying 450 MW.

Unit 4 automatically reduced power on load rejection. A Class II inverter failed, shutting off power to several control room indicators and controls. The control room operator, in consultation with the control room shift supervisor, decided to manually shut down the reactor using SDS 1.

The station's four standby generators automatically started. Two were used to supply Class III power to the station; two were left idling and available, but not connected.

D.10.5 Bruce A

SDS 1 was manually tripped on Units 3 and 4 as per operating procedures for a loss of Class IV power. SDS 1 was re-poised on both units when the station power supplies were stabilized.

The emergency transfer system functioned as per design, with the Class III standby generators supplying station electrical loads. The new qualified diesel generators received a start signal and were available to supply emergency loads, if necessary.

D.10.6 Bruce B

Following the loss of the electrical grid, the power level of all four Bruce B reactor units was automatically reduced.

Unit 6 experienced a SDS 1 trip while withdrawing adjuster rods to offset the xenon poison transient. One of the adjuster rods could not be automatically removed from the core due to a malfunction of the position feedback indicator. When Unit 6 tripped, Class IV power was lost. The emergency transfer system restored power to the unit.

Units 5, 7 and 8 were synchronized to the grid as soon as it became available.

D.10.7 Point Lepreau

A significant reversal of power flow on a transmission interconnection between New England and New Brunswick occurred during the power interruption. Point Lepreau rapidly dropped power by about 140 MW to match demand and remained operational, supplying loads in New Brunswick. The plant was operated in quiet mode for several hours.

D.10.8 Gentilly-2

The Hydro-Québec grid was not affected and Gentilly-2 continued to operate normally.

D.11 Pickering A Restart (Reference CMD 03-M56)

Unit 4 was synchronized to the grid on August 21. OPG raised power from 30% to a 60% hold point on August 26.

D.12 Bruce A Restart (Reference CMD 03-M56)

Following the successful completion of the final commissioning and safety system tests, Unit 4 was removed from GSS and achieved criticality in late August.

D.13 Discovery of Crack on Feeder PO9A at Point Lepreau

D.13.1 Description of Development (Reference CMD 03-M56.A)

On September 14, New Brunswick Power (NB Power) staff discovered a crack on reactor outlet feeder pipe P-09A during scheduled inspections during a planned maintenance and inspection outage.

Closer inspection of feeder pipe P-09A established the crack length to be between 15mm and 20mm on the pipe inside surface. The depth was approximately 50% through the wall; thus, the crack was detected prior to any leakage. It was the third time feeder P-09A had been inspected; the two previous inspections in 2001 and 2002 revealed no abnormalities.

NB Power was confident the inspection results showed that the P-09A outlet feeder crack was similar to the type previously observed at Point Lepreau on outlet feeders S-08A, K-16A, Q-08A and U-15C. The indication was in the same orientation and location on the first outlet bend. The response of the indication to tests was similar to the response from the sample feeder with an embedded crack used for qualification.

NB Power committed to repair feeder P09A and widen the scope of the inspection activities so that 100% of the outlet feeder first bends would be inspected before the unit was returned to service. A destructive examination of the removed section of feeder pipe P09A was to be conducted following the outage.

D.13.2 Follow-up

Further inspection during the outage led to the discovery of two more cracks. The work done during the outage, in conjunction with ongoing implementation of Point Lepreau's feeder cracking management strategy, was sufficient for CNSC staff to approve restart of the reactor. The Point Lepreau strategy uses the most recent information from the CANDU Owners' Group feeder integrity joint project, industry experience, and continuing inspection and surveillance of feeders.

D.14 Degradation of Bruce B Unit 8 Steam Generator Tube Support Plates

D.14.1 Description of Development (Reference CMD 03-M64)

Degradation of the seventh tube support plate was discovered during inspection of steam generators B03, B04 and B08 in Unit 8. The degradation was in the form of metal loss of the ligaments between the tubes at the periphery of the plate. The loss varied from minor to a complete loss of the ligament and appeared to be due to flow accelerated corrosion (FAC). The loss of support potentially increases the risk of tube damage due to vibration.

The most recent inspection results on all the steam generators in Units 3, 4, 5, 6, and 7 were reviewed. No indication of metal loss of the support plates was found. Steam generators B01, B02, B05, B06 and B07 of Unit 8 also showed no signs of metal loss.

Bruce Power began an investigation to determine the cause of the degradation and the required corrective actions.

D.14.2 Initial Follow-up (Reference CMD 04-M4)

Bruce Power performed extensive inspections of all tube support plates of the eight steam generators in Unit 8. Additional degradation was found in steam generator B08 on the fifth and sixth support plates. No degradation was found in the remaining Unit 8 steam generators or in any of the steam generators in the other Bruce B units. Anti-vibration support bars have been installed at the seventh support plate in Unit 8 steam generators B03, B04 and B08. All tubes in the degraded areas of the fifth and sixth support plates have been plugged, thus taking them out of service.

D.14.3 Additional Follow-up

Bruce Power prepared fitness-for-service reports for the steam generators. The steam generator dispositions were approved by CNSC staff on January 25th, 2004, allowing Unit 8 to return to service. CNSC staff still expects Bruce Power to provide a finalized root cause investigation report and strategy for monitoring and managing this type of degradation in the long term.

D.15 Pre-emptive Shutdown of Pickering B Unit 7 in Response to Algae Build-up

D.15.1 Description of Development (Reference CMD 03-M64)

The severe weather resulting from hurricane Isabel on September 19 resulted in an algae run in the Pickering B screenhouse. Unit 7 was proactively shut down to reduce the cooling water load and avoid tripping multiple units. Within two days, the threat of an algae run was reduced sufficiently to allow Unit 7 to return to power.

D.15.2 Follow-up

OPG upgraded equipment in the screenhouse to reduce vulnerability to this type of event. CNSC staff considers this to be appropriate.

D.16 Failure of Liquid Relief Valve at Pickering A Unit 4

D.16.1 Description of Development (Reference CMD 03-M64.A)

On November 13, a liquid relief valve in the PHT system failed open, discharging heavy water into the bleed condenser. The failure led to a reactor trip on low PHT pressure. All safety systems operated as expected and the reactor was confirmed to be safely shut down and placed in GSS. There was no loss of inventory from the bleed condenser.

The incident was caused by the failure of the diaphragm in the actuator of the relief valve. Unit 4 remained in GSS while the diaphragms of all four liquid relief valves were replaced and tested. The unit was scheduled to return to service on November 28.

D.16.2 Follow-up

OPG performed a thorough investigation of the cause of the valve failure. The results showed there were deficiencies by the manufacturer, mainly with the design and qualification of the valve actuator. The actuators had been replaced during the Unit 4 refurbishment with an environmentally qualified model. OPG replaced the diaphragms with a different material that meets the necessary environmental qualification (EQ) requirements. OPG also inspected all 35 similar valves (from the same manufacturer) that had been replaced during the refurbishment to ensure against this type of failure. The corrective actions identified by OPG also included a quality assurance (QA) audit of the manufacturer’s EQ program. CNSC staff considers OPG’s corrective actions to be comprehensive and appropriate.

D.17 Leakage of Feeder Weld Repair at Gentilly-2

D.17.1 Description of Development (Reference CMD 04-M4)

In early June, an increased tritium level in the reactor building was noted at Gentilly-2. The leak was identified to be coming from the weld between the upper and lower parts of outlet feeder G-09. The leak rate was monitored; it was consistently far below the operating threshold limit (20 kg/h) imposed by the operating policies and principles for the station. The initial leak rate was estimated to be in the order of 1.5 kg/h. It decreased to less than 0.5 kg/h at the time of the shutdown on August 31.

A subsequent metallurgical examination revealed that the cracking mechanism was consistent with inter-granular stress corrosion cracking (SCC) and that the crack was active during reactor operation within the repaired part of the weld.

Hydro-Québec repaired the failed feeder and, at the request of CNSC staff, enlarged the original scope of feeder inspections to include all feeder hub welds (repaired and non-repaired) considered significant for cracking. No additional cracked welds were found.

Subsequently, on December 1[1], OPG, Bruce Power and NB Power were notified about the potential existence of SCC at repaired welds and were requested to provide information on:

a) number, location and function (inlet or outlet) of any repaired welds on feeders;
b) safety case for continuing operation with affected feeders;
c) surveillance methods proposed (with schedule) to ascertain that SCC is not present in any affected feeders; and
d) any preventive or predictive measures or programs that the facility may have in place to deal with SCC.

D.17.2 Follow-up

Bruce Power responded to the CNSC with the requested information, indicating that baseline feeder bend cracking inspections were completed with no indications of SCC found in any unit. NB Power also provided the requested information, and indicated the intention of the industry to meet with CNSC staff to discuss ongoing work and plans to manage the risk of SCC in feeder weld repairs. OPG's studies have indicated that their limits and leakage monitoring procedures are adequate to detect leaks and shut down in a timely manner. OPG inspected several hundred outlet feeder bends (cold rolled) and no cracks were detected. The program will be expanded to include second tight-radius bends with low bend angles. Weld repairs, where accessible, will be done in 2004. Darlington plans to implement improved feeder cabinet leak data analysis software in 2004. OPG is also assessing alterative leak detection concepts.

D.18 Darlington Unit 2 Return to Service—Trip and Annulus Gas System Incident

D.18.1 Description of Development and Initial Follow-up (Reference CMD 04-M4)

On November 23, Darlington Unit 2 tripped from 30% full power on SDS 1 and 2 while it was returning to service from the station containment outage. The unit turbine tripped and the automatic transfer to back-up electrical supply did not operate properly resulting in a PHT pump trip and subsequent shutdown.

OPG performed a root cause analysis and found that incorrect valve positions, improper post maintenance checks and failed equipment contributed to this event.

On November 26, Darlington Unit 2 was again returning to service. Checks by Darlington operating staff, prior to start-up, failed to detect that the annulus gas system instrumentation was not available to provide PHT leak detection. Additionally, operations staff did not follow proper operating manual, abnormal incident manual or S-99 reporting procedures. The unit operated in this condition for three days.

D.18.2 Additional Follow-up to Trip at 30% Power

OPG completed root cause investigations and implemented mechanical and procedural actions. Human performance issues are to be addressed. CNSC staff met with senior station management and will follow-up during the next outage in March 2004.

[1] Original SDR erroneously says January 12, 2004.

D.18.3 Follow-up to Annulus Gas System Incident

OPG completed a root cause analysis. Corrective actions and procedural updates are being implemented and verified by CNSC site staff. Improvements to instrument indication will be completed as unit conditions allow (target of June 30, 2004), and site staff will follow-up on installation. A suitably revised training program will start on June 30, 2004 and site staff will attend.

APPENDIX E
GENERIC ACTION ITEMS

E.1 GAI 88G02—Hydrogen Behaviour in CANDU Nuclear Generating Stations

Loss of coolant accidents (LOCAs) can lead to substantial H_2 releases to containment over the long term from radiolysis (by radiation fields from intact fuel in the core) of the water in the calandria.

Also, for LOCA scenarios where the emergency core coolant (ECC) system is impaired, H_2 is released over the short term from oxidation of the over-heated fuel sheath by steam, and radionuclides are released due to sheath failures. Appreciable H_2 is also produced over the long term by radiolysis of the water in the sumps and calandria by the dissolved radionuclides from failed fuel. Sensitivity studies of post-blow-down steam flows through the core have indicated escalation of the H_2 and radionuclide releases for flows larger than zero but less than 100 g/s per channel, with a peak around 10 – 20 g/s per channel. The more significant long term H_2 releases have been shown to induce flammable and potentially explosive gas mixtures covering entire containment compartments. The short term H_2 releases can have similar local impact in certain regions of the affected compartments.

A significant safety issue, unless appropriate mitigation is provided, is the challenge posed to the integrity of the containment systems and the necessary or credited post-accident structures, systems and components (SSCs) inside containment, by the large combustion and potentially explosive loads from potential ignition of the long term H_2 releases. A second significant safety issue is related to the challenge posed to the post-accident performance of containment and its necessary/credited SSCs, by inadequate environmental qualification (EQ) to the induced harsh radiological and potential combustion conditions. Mitigation of the long term H_2 releases is also needed for viable severe accident management (SAM).

In CMD 99-121, CNSC staff expressed concerns about the completeness of the assessments of the transient H_2 distributions induced by the short term H_2 releases in containment. Detailed plans and schedules for resolution of all H_2 safety related issues, including those related to the long term releases, were requested of the licensees.

E.1.1 Hydro-Québec

Hydro-Québec has assessed the long term transient H_2 distributions in containment using the industry standard toolset (IST) lumped parameter computer code GOTHIC. A LOCA with loss of ECC (LOECC), and bounding post-blow-down core flows, was analysed with a passive autocatalytic recombiner (PAR) system in place. The results firmly established the effectiveness of the PARs in mitigating the combustion behaviour of the long term H_2 distributions. PARs were also observed to reduce the peak short term H_2 concentrations. To resolve the concerns of CNSC staff on the radiological source term, and to address issues related to the PAR self-start threshold, Hydro-Québec opted for a multi-pronged approach. It was decided:

1) to re-assess the short term post-blow-down H_2 and radiological releases with IST computer codes instead of the conservative CHAN code; and

2) to modify the PAR model of GOTHIC to incorporate deterioration of the self-start threshold; and
3) to collaborate with other utilities in the development of a probabilistic approach to address the releases from severe accidents and design basis accidents (DBAs).

In-situ testing of two PARs in the containment of Gentilly-2 and an in depth re-examination of the applicability of United States Nuclear Regulatory Commission (USNRC) combustion test results for EQ purposes are also underway. In addition, Hydro-Québec is collaborating with other utilities in the development of a probabilistic approach to address the releases from severe accidents and DBAs.

CNSC staff is greatly encouraged by the steps taken by Hydro-Québec to resolve the H_2-related issues, in particular those involving mitigation of the long term H_2 releases that pose the greatest challenge to safety. Staff is currently awaiting further details on the new methodology and on the plans and schedules for closure of the remaining issues related to the long term H_2 releases.

Hydro-Québec's safety assessments and efforts related to the implementation of measures to mitigate the long term H_2 releases for DBAs are judged to meet staff's expectations.

E.1.2 New Brunswick Power

In its analysis, New Brunswick Power (NB Power) considered LOCA with LOECC with bounding post-blow-down core flows. The calculated long term H_2 releases due to water radiolysis by the radionuclides from the failed fuel were found to be substantial (560 kg over 90 days). It was subsequently decided to install PARs to remove the H_2 releases. However, following the in-situ test of two PARs and the observed deterioration of the self-start threshold of the PAR located in the non-accessible area, NB Power postponed the PAR installation indefinitely. The need to maintain or modify the PAR system was claimed as justification. The option of replacing a single PAR element on a routine basis in the accessible areas was judged to be reasonable by NB Power but, to date, has not been acted upon.

NB Power claimed that over-pressurization of the containment over the long term due to ineffective cooling by the local air coolers (LAC) with fans and ducting damaged by potential long term H_2 burn loads, is not an issue. Post-accident venting for LOCA events was disallowed to avoid damage to the radionuclide filters by potential H_2 burn loads. EQ of necessary/credited post-accident SSC was to be carried out to the radiological conditions calculated for the plant's original design. Deflagration loads were not to be considered in the EQ, but the containment function was to be protected.

Based on rudimentary analysis of the short term releases, NB Power has concluded that loss of the LAC in the vault would not affect H_2 mixing in containment. Recently, assessments with a more appropriate 24 node 1D/3D GOTHIC model with fully functional LAC heat sinks have been submitted. NB Power also plans to evaluate the potential for explosive fast flames and transition to deflagration-to-detonation (DDT). NB Power is also participating with the other utilities in the development of a probabilistic approach to address the releases from severe accidents and DBAs.

CNSC staff finds the progress in assessments of the transient H_2 distributions encouraging. However, the long term H_2 releases have been underestimated, since the evaluation of the contribution from radiolysis of the water in the calandria by the radiation fields from intact fuel was not carried out. Also, the requested calculations of the PAR system effectiveness in mitigating the combustion behaviour of the long term H_2 distributions were not provided. Moreover, the perceived benign challenge to safety from potential post-accident H_2 combustion to the integrity of necessary and credited post-accident SSC is mostly based on the premise that PARs are in place. As noted above, installation of the PARs was postponed indefinitely, in spite of CNSC staff's request to proceed in view of their perceived net benefit.

CNSC staff therefore believes that NB Power should give priority to resolving the remaining issues related to the long term H_2 releases and to installing PARs, at least in the accessible areas of containment. This is needed to firmly establish the integrity and function of the necessary/credited post accident SSC.

NB Power's safety assessments and the efforts related to the implementation of measures for mitigation of the long term H_2 releases for DBAs do not meet staff's expectations.

E.1.3 Bruce Power and OPG

Bruce Power and OPG have opted for a two-pronged approach for resolution of the H_2 related issues:

1) a probabilistic assessment of the likelihood of LOCA and ECC impairments involving channel flow in the critical range; and
2) the adoption of a new, realistically conservative methodology for re-assessment of the radionuclide and H_2 source terms.

Bruce Power and OPG have argued that, based on the new evaluations, the scenarios with critical post-blow-down steam flows belong in the severe accident category. As a result, these licensees took the decision to consider the source terms from only a single LOCA event in their EQ programs To achieve closure of the remaining H_2 issues of GAI 88G02 (associated with credible scenarios), the use of more realistic assumptions has been proposed. Closure of the remaining H_2 issues related to severe LOCA and LOECC sequences is to be addressed by SAM. PARs might be considered by Bruce Power and OPG to remove the H_2 produced by credible and severe accidents, subject to:

1) the results from in-situ tests of the PAR self-start threshold to be completed two months after Bruce A restart; and
2) the definition of realistic conditions for self-start in the design and performance requirements for PARs to be adopted by Bruce Power and OPG.

CNSC staff believes, however, that the conclusions from the above overall approach are premature. CNSC staff has reviewed the proposed methodology for resolution of the issue and has identified a number of concerns. Of particular importance is the severity of the consequences of scenarios with critical post-blow-down core flows and the lack of completeness and robustness of the probabilistic analysis used to classify these scenarios as severe accidents. Concerns about the completeness and/or conservatism of assessments of the radiological and H_2 source terms with the new realistically conservative methodology and on assessments of their subsequent transient distribution in containment for credible DBAs were also expressed.

At this time, it is not clear to CNSC staff whether the licensees' adopted course of action will be sufficient to resolve this issue.

CNSC staff has met with representatives of Bruce Power and OPG and is currently preparing a letter outlining the path forward for the long overdue resolution of remaining concerns on the submitted probabilistic safety assessments that challenge the basic premises of their conclusions. Another meeting is planned to discuss the appropriate assessment of the source terms and of their transient distributions for DBAs.

For Bruce Power and OPG, the safety assessments and the efforts related to the implementation of measures for mitigation of the long term H_2 releases for credible DBAs do not meet CNSC staff's expectations.

E.2 GAI 90G02—Core Cooling in the Absence of Forced Flow

Failure of the primary heat transport pumps to provide forced circulation of water for fuel cooling is a possibility in some accident sequences. The reactors then rely on natural circulation of the coolant to remove residual heat from the fuel to the steam generators. Natural circulation experiments done at Atomic Energy of Canada Limited's (AECL) Whiteshell Laboratories showed degraded cooling in some channels if coolant inventory is low. The experimental results cast doubt on the safety analysis predictions regarding the effectiveness of natural circulation under partial inventory conditions. Licensees were requested to identify the causes leading to the observed degraded cooling conditions and, if needed, to revise their safety analyses or implement design changes.

This GAI has been closed for all licensees except NB Power. Its progress has been slower than that of the other licensees because, early on, it adopted a very ambitious analytical strategy to resolve the issue. CNSC staff did not accept the results of that approach because there was insufficient experimental verification.

NB Power subsequently adopted an approach more in line with the rest of the industry. It has made a number of submissions addressing the closure criteria for the GAI and has requested closure. CNSC staff has assigned low priority to the review of this work since the expectation is that the Point Lepreau reactor will behave in a very similar way to other CANDUs and so the safety significance is low. NB Power's performance on this GAI has been satisfactory.

E.3 GAI 90G03—Assurance of Continued Nuclear Generating Station Safety

Safety-related functions in nuclear power plants must remain effective throughout the life of the plants. GAI 90G03 was opened to ensure that licensees implement programs to prevent, detect and correct any significant degradation in the effectiveness of safety-related functions. This GAI was closed for all licensees in 2003, on the basis that the CNSC will continue to adequately monitor aging management through licensing and compliance activities.

E.4 GAI 91G01—Post-Accident Filter Effectiveness

In certain postulated accidents, venting of containment may be needed to reduce the risk of an uncontrolled release of radioactive material. The licensees have been required to demonstrate that the filters are capable of performing their design function and that adequate testing and maintenance activities for them are in place.

The GAI is already closed for Hydro-Québec. CNSC staff is waiting for NB Power to present a similar argument to Hydro-Québec and to provide details on how conditions in containment would be stabilized in the long term following such accidents.

A meeting was held in February 2003 between Bruce Power, OPG and CNSC staff to discuss the progress of the work and the requirements for meeting the closure criteria for GAI 91G01. At this meeting, it was also agreed that the impact of hydrogen burns on emergency filtered air discharge system (EFADS) performance will be addressed as part of GAI 88G02, and that the issues related to qualification of filters and components will be addressed in the framework of the licensees' EQ programs. This signalled the completion of Phase II (EFADS filters) of the work for Bruce Power and OPG for this GAI.

In October 2003, Bruce Power and OPG submitted reports for completion of the Phase III (non-EFADS filters) and requested closure of GAI 91G01 for Bruce B and the OPG plants respectively. Bruce Power also submitted a work plan for Bruce A. A meeting was held in December 2003 between Bruce Power, OPG and CNSC staff to discuss these submissions. Additional information was requested from the licensees to ensure consistency between the assumptions in safety analyses and the operation mode of the non-EFADS filters.

Overall, CNSC staff is satisfied with the progress towards resolution of this GAI.

E.5 GAI 94G01—Best Effort Analysis of ECCS Effectiveness

This GAI, only applicable to the multi-unit stations, was originally opened to establish adequate validation of improved methods/codes used to predict the effectiveness of the ECC system during a large LOCA. These predictions were considered necessary to supplement the bounding calculations which demonstrated that radiological dose consequences were acceptable. Since then, another GAI was opened to cover validation of all computer codes used in safety analysis (GAI 98G02). In that sense, the newer GAI superseded GAI 94G01. Following the closure of GAI 98G02 for Bruce Power and OPG, GAI 94G01 was closed for the multi-unit stations in early 2004.

E.6 GAI 94G02—Impact of Fuel Bundle Condition on Reactor Safety

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 the CANLUB layer, oxidation of defective fuel and fission product release.

Fuel bundle degradation depends on the reactor, fuel channel and fuel designs, fuel manufacture 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. In addition, fuel bundle degradation is sometimes also accompanied by fretting and scratching of the pressure tube and may depend on other phenomena such as pressure tube creep.

The effects of bundle degradation on reactor safety are not fully known, partially because of limited experimental data and safety analysis methods. As such, it is important to monitor fuel performance by conducting fuel inspections and examinations, and integrated evaluation of relevant information. As such, the important fuel and fuel channel parameter to measure are not known. Although some fuel inspections have been conducted and the results have been 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.

Consequently, the licensees have been required to:

- implement an action plan to eliminate excess fuel and fuel channel degradation in acoustically active channels; and
- implement an effective, formal, and systematic process for integrating fuel design, fuel and channel inspection (in-situ), fuel and fuel channel laboratory examination, research, operating limits and safety analysis.

This GAI was closed for OPG and Bruce Power in 2001 and 2002 respectively. In April 2002, Hydro-Québec made a submission to the CNSC requesting the closure of this GAI. CNSC staff reviewed the submitted documents and issued an evaluation report in July 2003. The GAI closing for Hydro-Québec is pending their response to CNSC concerns. As for NB Power, their progress has been slow and the CNSC has not received a request for closure. NB Power is developing the necessary policies and procedures, and is expected to submit them to the CNSC in the first half of 2004.

E.7 GAI 95G01—Molten Fuel / Moderator Interaction

A severe flow blockage in a fuel channel, or an inlet feeder stagnation break, could potentially lead to fuel melting, channel rupture and ejection of molten fuel into the moderator. It is uncertain as to whether the resulting molten fuel/moderator interaction could damage the shut-off rod guide tubes and prevent SDS1 from functioning properly. It could also damage other fuel channels, or the calandria vessel itself.

There has been a long-standing difference of opinion between CNSC staff and licensees and their respective consultants on the severity of the molten fuel/moderator interaction. Starting the first quarter of 2000, however, licensees initiated an experimental program to resolve this matter. A panel of three independent fuel-coolant interaction experts was set up to review the experimental program and the resolution criteria proposed by industry. CNSC staff accepted the panel's final recommendations and the industry's proposed closure criteria.

CNSC staff has also accepted the licensees' proposed experimental program schedule, which plans to conclude the experimental program by the third quarter of 2005. At that time, CNSC staff expects a submission from the licensees, with the experimental results, and a request to close this GAI. However, the project has encountered some unexpected technical challenges. Although they are now mostly resolved, some preparatory tests have been delayed by nine months; this is in addition to a delay in obtaining the code classification approval for the test facility. CNSC staff views the overall progress of this issue as slow, but considers the current plans acceptable.

E.8 GAI 95G02—Pressure Tube Failure with Consequential Loss of Moderator

Traditionally, the single and dual failure concept in safety analyses calls for analyses of initiating events, plus analyses of initiating events coupled with failure of one of the special safety systems. For the postulated scenario of LOCA plus LOECC, the moderator system has been credited in the analysis as a heat sink. Heat transfer to the moderator is assumed to be via pressure tube contact with calandria tubes following pressure tube deformation due to heat-up. This mode of heat transfer has been accepted by CNSC staff, since the moderator was considered to be independent of postulated initiating events and ECC failures. However, experiments suggest it is possible for the moderator water to drain during the following postulated scenario: rupture of the pressure tube and then end-fitting bellows, followed by calandria tube failure, guillotine failure of the already ruptured pressure tube, end fitting ejection and drainage of the moderator. This postulated event could result in severe damage to a large number of channels, with consequences in excess of those anticipated in the safety report.

In a position statement addressing this GAI, licensees were requested to provide acceptable proposals for a course of action, including possible design changes to be implemented by the end of 2000 that would result in the mitigation of, or at least a significant reduction in, the impact of the consequences of such an event.

An industry plan of action was submitted to CNSC staff in May 2000. In this plan, the industry presented its proposed evaluation criteria, including a proposed cost-benefit methodology. Subsequently, CNSC staff has modified its position statement to refer to the CNSC policy on the use of cost-benefit arguments, and to modify the closure criteria and the completion schedule to reflect recent CNSC staff and industry discussions.

The industry has submitted the basis for their plans of actions in accordance with the revised position statement for this GAI, and requested closure. Assessment of this submission is on hold, pending the finalization of the guidelines for the use of cost-benefit analysis.

E.9 GAI 95G04—Positive Void Reactivity Uncertainty – Treatment in Large LOCA Analysis

Accuracy of void reactivity calculations is a significant safety issue in the analyses of DBAs involving channel voiding, especially for large LOCAs (LLOCAs). In 1995, CNSC staff raised concerns about the adequacy of available evidence in support of best-estimate predictions of void reactivity, and subsequently requested all licensees to complete a suitable experimental program to improve related safety analyses, and to undertake adequate interim measures.

In 2001, a CANDU Owner’s Group report on void reactivity error assessment for CANDU reactors was issued. It summarized the results arising from the overall industry program to address GAI 95G04. It was concluded that the new industry standard toolset (IST) reactor physics suite of computer codes over-predicts the void reactivity of CANDU fuel when compared to the ZED-2 research reactor measurements. The report recommended fuel-type specific values for the errors to be applied in void reactivity calculations by IST reactor physics codes for operating CANDU conditions at all fuel burn-ups. This recommended value of over-prediction of void reactivity has been credited in the recent LLOCA safety analyses with the new IST reactor physics suite of codes.

The acceptability of the estimate of uncertainty in the IST reactor physics codes' prediction of void reactivity for operating CANDU conditions has also been discussed in an industry-proposed independent panel assessment. The panel report was completed and issued in January 2003. The industry dispositioned the recommendations that were made and proposed relevant research and development activities. The CNSC continues to review options to address remaining issues.

E.10 GAI 95G05—Moderator Temperature Predictions

In some LLOCA events, the integrity of fuel channels depends on the capability of the moderator to act as the ultimate heat sink. As fuel channels heat up, pressure tubes (PTs) balloon radially and make contact with the calandria tubes (CTs). Fuel channels remain intact upon contact if the moderator fluid outside the CT is cold enough to provide good heat removal capability. Channels may fail, however, if the moderator temperature is too high to prevent the outside of the CT from drying out following contact on the inside with the PT.

In view of the severe consequences of channel failures, and the small safety margins that currently exist with respect to moderator temperature (or moderator subcooling) requirements, CNSC staff requested the validation of the computer code used to calculate the moderator temperature distribution against three-dimensional moderator tests. OPG submitted a test program in accordance with the CNSC position statement issued for this GAI. CNSC staff agreed to the OPG test program and schedule, and has held regular meetings with OPG staff to both monitor the progress made in this work and to provide feedback.

The experimental program was completed in December 2001 and the code qualification documents, including an interim validation report, have been submitted to CNSC. The code validation activity in conjunction with analysis of the data from the experimental program has been slow. However, in 2003, OPG and Bruce Power increased their efforts in this respect and completed work related to data analysis and a validation exercise related to Bruce A. In addition, a plan to complete the validation program was presented to CNSC staff in December 2003. Discussion on the scope of work needed to close this GAI is underway. As for NB Power and Hydro-Québec, their past participation in this GAI has been minimal, even though computer codes related to moderator temperature prediction also need to be validated for CANDU 6 applications. However, in 2003, they joined OPG and Bruce Power to share the activities related to this GAI.

E.11 GAI 96G01—Fire Protection for CANDU NPPs

This GAI became inactive when remaining issues related to fire protection for licensees were transferred to station-specific action items. This GAI had already been closed for Hydro-Québec in 2000; it was closed for OPG, Bruce Power, and NB Power in 2003.

E.12 GAI 98G01—PHT Pump Operation under Two-Phase Flow Conditions

The operation of the primary heat transport (PHT) pumps under LOCA conditions can be detrimental to the integrity of the system piping due to the generation of large pressure pulsations and excessive pump vibration. In the past, a piping analysis was done using limited experimental information from laboratory tests. This approach was very sensitive to the interpretation of the test data and their application to the reactor. Re-assessment was needed to obtain a more realistic representation of the behaviour of the pump and piping under various accident conditions. In particular, the fatigue analysis of the PHT system piping required updating with the use of a conservative forcing function.

This GAI was closed for Bruce B, Pickering, Darlington, and Gentilly-2 prior to 2003. It was closed for Point Lepreau in 2003 following review of their analysis of pipe fatigue in the PHT system when subjected to pump-generated pressure pulsations caused by two-phase flow conditions.

The results of a re-assessment of the integrity of Bruce A piping when PHT pumps operate under two-phase conditions were reviewed by CNSC staff in 2003. Additional confirmatory piping stress analyses were requested from Bruce Power. Following their submission, Bruce Power's request for closure of this GAI can be considered.

E.13 GAI 98G02—Validation of Computer Programs Used in Safety Analysis of Power Reactors

In the past, CNSC staff assessed licensees' computer programs and safety analysis methods, and identified several inadequate practices with respect to computer program validation. Examples of poor practices include lack of a managed process in performing validation, poor documentation of computer program validation, poor applicability of validation due to the limited range of conditions in the validation experiments in comparison with the reactor analysis, and inadequate assessment of the impact of dimensional scaling and important phenomena for which adequate validation data do not exist. CNSC staff concluded that these inadequate practices eroded overall confidence in the safety analyses results.

The industry has responded to this GAI favourably by establishing a quality control process to improve the computer code validation, and by achieving an overall level of baseline validation for a specific set of major computer codes used in safety analyses. These efforts, once confirmed, are considered to be sufficient to warrant the closure of this GAI, and it has been closed for Bruce Power and OPG. An audit of NB was carried out in 2003, and a similar audit is planned for Hydro-Québec in 2004. Evaluation of the results will be followed by a decision regarding closure of this GAI for those two licensees.

E.14 GAI 99G01—Quality Assurance of Safety Analysis

The CNSC expects power reactor licensees to conduct operations in accordance with a quality assurance program. This program includes requirements for various safety-related activities, including safety analyses. The acceptability of the safety-related information established by the safety analyses depends on the degree of conservatism incorporated into the analyses. It also relies on the credibility of the analytical tools and activities (such as computer codes, methods and input information). Licensees need to perform safety analyses in a systematic manner, using quality assurance principles, to ensure confidence in the licensing basis and safe operating envelope for each facility.

In recent years, CNSC staff had become aware of an increasing number of occurrences of poor safety analysis practices by power reactor licensees caused by inadequate quality assurance. These poor practices were identified through audits and assessments. In 1999, staff's conclusion that inadequate quality assurance of safety analyses was resulting in a reduction in the overall confidence in the safety analysis results led to the initiation of this GAI.

The industry has responded by establishing quality assurance frameworks and procedures related to safety analysis, and by taking actions to satisfy all relevant closure criteria. This GAI has been closed for Bruce Power, and is under review for other licensees. The results of recent audits of OPG and NB Power, and an audit planned for Hydro-Québec in 2004, will be considered prior to considering closure of this GAI for those licensees.

E.15 GAI 99G02—Replacement of Reactor Physics Computer Codes Used in Safety Analysis of CANDU Reactors

Licensees use reactor physics methods and computer codes to support nuclear design, operation and compliance with the safe operating envelope. There are stringent requirements on accuracy and validation of these methods and codes due to their role in the confirmation of safe operation. Recent experimental data, as well as reviews of key computer codes, identified several shortcomings. These deficiencies are related to inaccurate predictions of key parameters for accident conditions, lack of proper validation and a significant lag of licensees' methods and codes behind the current state of knowledge in this area. These shortcomings had a negative effect on the overall confidence in the results of reactor physics analyses, especially for those analyses where safety margins are small.

Under this GAI, licensees are required to carry out a structured program of replacement of reactor physics computer codes. In February 2001, an industry project to analyze a power pulse following a LLOCA with the new set of reactor physics codes resulted in the prediction of more severe consequences than those presented in earlier licensing submissions. To mitigate the potential effects of this, the licensees implemented more restrictive operating limits, such as flux tilt limit, moderator and coolant purity limits, and moderator poison load limit to compensate the increase in the predicted power pulse. Following imposition of those restrictions, licensees continued their structured programs to replace reactor physics computer codes.

A report of an independent expert panel (see GAI 95G04) addressed the acceptability of estimated uncertainties of key parameters predicted by the codes. Two licensees (Bruce Power and OPG) completed an agreed set of activities and declared the new reactor physics toolset in service for future accident analysis. Work is continuing on a second set of activities on code validation. The work of NB Power and Hydro-Québec is behind schedule.

E.16 GAI 00G01—Channel Voiding During a Large LOCA

CNSC staff has a concern that the computer codes used for prediction of overpower transients for CANDU reactors with a positive coolant void reactivity coefficient have not been adequately validated. This GAI requires the licensees to carry out direct void fraction measurements, provide an assessment of the scaling of the results to the phenomena expected in the reactor, perform validation exercises using these data and complete an impact assessment on the safety margins.

Tests with void fraction measurements in AECL's RD-14M facility have been completed, and data analysis reports have been submitted to the CNSC. The industry has provided information on the computer code validation exercises and the scaling assessment.

There are two outstanding issues before closure of this GAI can be considered. One is the satisfactory resolution of the scaling issue, which is related to appropriate scaling of the test results to reactor conditions. The licensees continued to work in 2003 to incorporate CNSC staff comments in their assessment. The second issue is the impact of uncertainty that results from the computer code validation. It has been indicated that the licensees would address this issue through either a "best estimate and uncertainty" analysis or via a re-analysis of the LLOCA scenario. CNSC staff is in discussion with the industry as to how the impact on the safety case can be adequately addressed.

E.17 GAI 01G01—Fuel Management and Surveillance Software Upgrade

This GAI was initiated as a follow-up to the closure of GAI 95G03. The GAI only relates to Bruce Power and OPG.

Compliance with reactor physics safety limits that define the safe operating envelope, such as channel and bundle power limits, is based on analyses performed with a fuel management computer code. Recent, more rigorous scrutiny of the accuracy of methods, acceptance criteria, assumptions and results of safety analyses of various DBAs led to significant restrictions of operating parameters, including channel and bundle powers, and introduction of additional physics parameters for compliance purposes, such as fuel string relocation reactivity and minimum margin to axial constraint. As such, the significance of compliance with safety-related reactor physics limits has increased. This has enhanced the need for an improved analytical model, validated over a broader range of applications and conditions as well as better-defined compliance allowances and more consistent procedures.

To achieve closure of this GAI, licensees were required to undertake a structured program for reactor core surveillance that covers the fuel management software upgrade and validation as well as validation and qualification of the error compliance methodology.

Commensurate progress has been made so far. Bruce Power and OPG submitted detailed work plans and schedules, as well as semi-annual progress reports. Work is divided into two main phases. Phase I deals with modeling improvements to the SORO computer code and Phase II deals with estimation of error allowances.

A significant milestone was achieved in 2003 with the implementation of a first-improved version of the computer code WIMS-IST-SORO. CNSC staff is closely monitoring the progress of this GAI.