Canadian Nuclear Safety Commission
Commission canadienne de sûreté nucléaire

Convention on Nuclear Safety Second Review Meeting – April 2002

Responses to Questions Presented to Canada

July 2002

Canada

	Country	CNS Article	Report Reference	Question	Answer
					licence cycle is not necessarily the most effective use of regulatory resources, nor is it consistent with best practices in other regulatory regimes. CNSC staff has recently recommended to the Commission that there be a transition to longer licence periods and has stated that, for power reactors, a program of routine Periodic Safety Reviews would be coupled with the longer licence period.
8	Germany	7.2 (i)	Section 7.3: p. 36-37	The license requirements include: "....proposals by the licensee for procedures, measures, programs, " etc." Are those based on the regulatory guidelines shown in Table 7.1? Regarding p. 10 last bullet "... more prescriptive regulations", is it intended to use more prescriptive regulations in future?	The licence application requirements for programs are set out in the Regulations issued under the NSC Act (section 7.2.1 of the report). For some areas, the documents in Table 7.1 provide further guidance or requirements on specific programs. In other areas, the CNSC uses national standards such as the CSA standards on Quality Assurance. On the non-prescriptive aspects of Canada's nuclear regulations, please see the first paragraph of the answer to Q7.
9	Japan	8	Annex 8.1: p. 210	It is reported that CNSC has approximately 450 employees. The present number of CNSC staff is larger than that of AECB, which was described in previous report, while CNSC succeeding AECB. 1) Which regulatory areas does CNSC strengthen? 2) Does CNSC increase number of staff further in the future?	There is no significance in the change in numbers of employees reported in the two reports. It merely reflects the normal change in staffing levels as people leave and are replaced. There has been an increase in the number of security advisors in response to the Sept. 11th events. There is no current intention to increase staffing levels further in the future. Refer to the response for Q2 for regulatory areas strengthened by the introduction of the new Act.
10	USA	10	N/A	The report does not appear to address how allegations of safety concerns are addressed. How are allegations of safety concerns addressed by the regulatory body?	The CNSC takes all allegations, whether made by members of the public or concerned workers, very seriously. All such allegations are investigated. The extent of the investigation depends on information received and the severity of the allegation. The *Nuclear Safety and Control Act* contains a provision that forbids employers from taking disciplinary action against people who, as part of their job, provide information to the Commission, or its staff.
11	Germany	7.2 (ii)	p. 38	Can affected individuals take legal action against licenses issued by CNSC?	Experience to date is that there have been few challenges and these were almost exclusively related to challenges on process, mostly to do with environmental reviews.

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12	Germany	10	Section 10: p. 53-56	The utilities have to provide, and after CNSC approval, have to follow the Operating Policies & Principles document to ensure operation of the station within the boundaries of the Safe Operating Envelope. Have violations of the OP & P been identified yet and what have been the consequences?	Non-compliances to the Operating Policies and Principles have occurred during the period covered by this report. Each non-compliance was assessed by CNSC staff to determine its safety significance and its safety consequence, and appropriate regulatory action taken.
13	Japan	7	Section 7.2: p.33	It is reported that the legislation, called "An Act Respecting the Long-Term Management of Nuclear Fuel Waste", will require nuclear utilities to form a waste management organization as a separate legal entity. 1) More information would be appreciated on purpose and duty of this organization. 2) Do utilities bear no responsibility for waste management after the new legal entity is established?	Information on the Act Respecting the Long-Term Management of Nuclear Fuel Waste is included as Attachment 2
MAINTAINING TECHNICAL CAPABILITY					
14	Germany		Introduction: p. 2-4	The report mentions Federal Government funding for research and development activities related to CANDU technology at the level of $ 100 m /year. Is it envisaged to keep this level constant in the years to come?	The annual $100 million appropriation to Atomic Energy of Canada Limited is subject to an ongoing review and approval process by the Government of Canada. Accordingly, it is impossible to predict appropriation levels in future years.
15	Hungary	Intro & 11	Intro & 11	Is there any separate regulatory R&D programme and if yes, what sources are available for it? How can the CNSC use the results of the R&D programme sponsored by CANDU Owners Group (COG)?	There is a separate CNSC research budget with a current annual budget of about $CAN 2.5M. The CNSC has access to the results of all COG R&D reports, many of which deal with work to respond to CNSC imposed actions. These results are used by the CNSC in formulating regulatory positions.
16	Japan	11	Section 11.6: p.63	It is reported that the industry is proposing the development of CANDU-specific technical Centers of Excellence, and utilities propose to collaborate with the R&D organization to ensure that appropriate succession planning is in-place. Could you	The concept of Centres of Excellence is still evolving. The CANDU Owners' Group (COG) has been asked to develop the concept for implementation for the COG cost-shared R&D program. The model that is currently under discussion is based on the use of a management team for each Centre of Excellence that comprises representatives from participants in the R&D program and from COG. Input would

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				provide more information for us to understand the concept of proposed organization, such as its budget, its number of staff personnel and the working period of the organization?	be provided by R&D suppliers and educational institutions, particularly on issues related to capability that is at threat, or to ensuring a supply of qualified graduates for the nuclear industry. The management team would ensure that there is a strategic plan for the technical area that ensures members needs are addressed, and that key capabilities are maintained.
17	Japan	11	Section 11.6: p. 63	It is reported that Canadian Nuclear utilities have proposed the establishment of a Network of Excellence in Nuclear Engineering at Canadian universities to address the issue of long-term capability maintenance and the CNSC is also contributing to this program. Could you provide more information for us to understand the role of the CNSC in attracting younger people?	The CNSC is providing a small financial contribution to the University Network of Excellence in Nuclear Engineering (UNENE). The CNSC will have a representative on the UNENE steering committee. The CNSC is interested in this program for two main reasons. First, the CNSC has need of highly trained engineers and scientists and the UNENE addresses this need in part (CNSC will be recruiting staff from graduates of the program). Second, the UNENE will provide a source for upgrading current CNSC staff members' knowledge levels. In addition to these direct benefits, the UNENE will have important benefits to the industry, not just in staff training, but in enhancing the research capability at Canadian universities. For these reasons, the CNSC has chosen to actively contribute to the program.
18	Germany		Introduction: p. 9	CNSC requested an evaluation of the research and development state. What kind of actions are recommended to avoid that "the support capability reaches a critical level"?	The CANDU Owners Group report identified a number of strategic recommendations: • The industry takes steps to ensure that the federal government maintains its support to the Canadian nuclear national laboratory. • The R&D supplier organizations and the utilities ensure that minimum R&D capability is maintained in critical areas and that strategies are defined and programs added as required to ensure that these capabilities remain available in the future. • In critical areas, additional funding be provided to ensure that staffing can be augmented to permit knowledge transfer and staff development. • Overall expenditure on R&D be augmented and stabilized so that an attractive environment for existing and potential R&D staff is put in place. In addition the CNSC has placed more stringent R&D reporting requirements on the licensees to enable stronger regulatory oversight

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					on changes to the R&D programs and funding. Despite the above, maintaining capabilities in the longer term remains a challenge.
ASSESSMENT AND VERIFICATION OF SAFETY – SAFETY ANALYSIS					
19	Romania	14	N/A	What arrangements are there to ensure that the station Safety Analysis Report is updated following plant modifications?	Through licence conditions, licensees are required to update their safety reports at regular intervals. These updates are expected to include the impact of plant modifications.
20	Japan	6	Section 6.1: p.16	It is reported that the requirements in C-006, Rev.1 represent an increase in both the scope and the rigor of design basis accident, and that these requirements have not been readily accepted by the licensee who have proposed an incremental approach to address the scope of the design basis accidents. How will CNSC and licensee compromise on this difference?	An exercise recently completed for Pickering A restart has demonstrated that the requirements of C-006Rev 1 can be met. It was found that the differences between the CNSC and the licensees can be accommodated.
21	Germany	18	Section 18.1: p. 111, 245	Are some/all plant operating parameters constituting the safe operating envelope kept within the required boundaries by automatic control (what time span is considered as sufficient)? Are there special provisions to keep the operators informed on actions of the automatic system?	Not all limits are under automatic control. For example, channel and bundle power limits are controlled by appropriate fuelling and reactor operation. Many limits are subject to automatic control within the safety analysis limits. It is normal for the control band to be more restrictive and for automatic alarms to be provided before the control band is exceeded.
22	Germany		Introduction: p. 6	For severe accidents, such as a LOCA combined with a loss of emergency core coolant injection, the pressure tubes will sag and/or strain into contact with the calandria tube where further deformation will be arrested by the cooling of the moderator system. Should channel failure occur (for example, due to a further equipment unavailability resulting in a loss of moderator heat removal), then such failures will be	Most of the assessment for severe accident scenarios is based on analytical models and there is little integral experimental data on the event sequence where large-scale core disassembly is likely to occur. Although the damage to the containment due to the massive failure of the pressure vessel at high pressure is not a concern, the potential for steam explosion and its impact on the containment boundary, as the degraded fuel channels progressively fall into the remaining liquid moderator, needs to be addressed.

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				spread out in time "softening" the load on the containment. Has this scenario been derived from integral experiments or by analytical means?	
23	Germany	14	Section 14.3.4: p. 83-84	Are multiple feeder pipe failures taken into account in corresponding analyses?	The safety analysis examines the loss of coolant accident scenario by assessing the whole spectrum of break sizes, from very small leak to guillotine rupture of the largest pipe (such as reactor inlet and outlet headers) in the heat transport system. Analysis of a single feeder pipe break is explicitly included in this spectrum. Multiple feeder pipe breaks are not explicitly included on the basis of low probability of occurrence.
24	France	18	Section 18.1	Does the design of nuclear power plants in Canada include some specific features relating to severe accident management in order to reduce the probability of large off-site releases requiring short-term off-site response (as indicated in INSAG 12)?	The situation is similar to that for LWRs (for example, Canadian plants are assessing the need for hydrogen recombiners). However, CANDU reactors have the advantage that most events sequences leading to core disassembly occur over a longer timeframe. The Canadian power reactor licensees are now working on the implementation of severe accident management guidelines.
25	Germany	17	Section 17.1: p. 109, 237	The criteria of site-related factors include "flight paths of major airports with the possibility of airplane crashes". Is the probability of an airplane crash considered here only, or are the plants designed to withstand such crashes? If so, what kind of load functions have been applied?	The design guides for the containment structure for some reactors included consideration of an impact of an external missile. The design of the previously built reactors was also assessed. These design requirements were based on the probability of the mass and the velocity of the impacting missile. This, in turn was a function of the distance from the airport and from the flight paths, air traffic density and the distribution of the size of the airplanes that fly overhead. A review of the adequacy of these requirements is currently underway.
26	Germany	14	Section 14.3.2: p. 80-81	It is stated that the reliability requirements for the special safety systems of the Regulatory Guides (R-7, R-8 and R-9) are specially referenced only in the Darlington NGS operating license. Are the unavailability targets for these and other safety related systems different for different plants? If so, will they be harmonised by the standard under development "Reliability Programs for NPP"?	All NPPs use the same reliability target as that given in Regulatory Guides R-7, R-8 and R-9. The older plants, notably Pickering NGS A, are allowed some latitude (e.g., $2 * 10^{-3}$ for ECCS) in meeting this target.

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ASSESSMENT AND VERIFICATION OF SAFETY – PERIODIC SAFETY REVIEW					
27	France	6	Section 6	From the report it appears that a continuous evolution of Safety analysis requirements of Candu nuclear plants took place up to the last document issued in September 1999. Could Canada indicate when performing a periodic safety review of an old plant, if the new safety analysis requirements are applied in order to identify the practicable upgrades of the plant?	Periodic safety review is not yet mandatory in Canada. However, a review equivalent to PSR was conducted for Pickering A restart. This review did reveal the need for a number of upgrades which are now being implemented (refer to Annex 14.2 of Canada's report).
28	Hungary	14	Section 14	How are the Periodic Safety Review requirements related to the regulatory requirements of lifetime extension? What kind of Regulatory Guides has been issued or planned to issue for life extension and operational license renewal?	The CNSC does not have formal regulatory requirements for life extension; however, the CNSC has required the equivalent of a PSR be done for the Pickering A and Bruce A restarts and the Point Lepreau refurbishment project.
29	Slovenia	14	Section 14.3.2: p. 84	Periodic Safety Review that would be required…Please explain what is the position of CNSC regarding PSR. Is it performed only as an extraordinary review because it is supposed that other regulatory requirements cover its scope sufficiently or should it be performed on a regular basis?	PSRs are not yet a regulatory requirement. Much of the information required by a PSR is however covered during the periodic licence renewal process and more information has been requested in support of reactors which are returning to service from lay-up.
30	China	14	N/A	How did Canadian Nuclear Safety Commission conduct the periodic safety review(PSR) for operating NPPs? Please give an example.	Refer to the responses to Q28 and Q29.
ASSESSMENT AND VERIFICATION OF SAFETY – PROBABILISTIC SAFETYASSESSMENT					
31	Germany	14	Section 14.3.2: p. 80-82	What level of PSA has been covered by existing analyses and those under preparation? Is it intended to make the performance of comprehensive plant-specific PSAs at regular intervals a regulatory requirement? If so, will they include low-power and shutdown states and a new	PSAs exist for Pickering A (PARA) and Bruce B (BBRA), and are under preparation for Pickering B (PBRA), and Darlington (DARA). These are essentially Level III PSAs that include internal events, flooding and also address all operating and shutdown states. A regulatory Policy is in preparation requiring an up to date Level II PSA for every NPP. In addition for new designs submitted for CNSC review (e.g., CANDU 9) a PSA was performed at the design stage.

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				evaluation of site-specific effects, external hazards and higher burn-up?	
32	France	17	Section 17.1	Could Canada give more information concerning the criteria used to define the different external events (natural and man made)? In particular, does Canada use a probabilistic objective?	The CNSC defines the external events to be considered for a specific NPP according to the physical location of the plant. Any credible external event is considered, without formal consideration of its probability. For instance, due to the proximity of a railway line, the consequences of an explosion in a train load are included in the external events set for Darlington.
33	France	18	Section 18.1	Canada indicates that an unavailability target of 10^{-3} year/year is set for safety systems. Does this apply to both shutdown systems or to the shutdown function as a whole? Are these two systems diversified or identical?	The unavailability target applies to each special safety system (shutdown system #1, shutdown system #2, ECC, containment) individually. Thus each shutdown system must have a demonstrated unavailability of $< 10^{-3}$ years/year without reliance on any other means of reactor shutdown, including the control system and its separate devices. The two shutdown systems are divers in terms of spatial separation, independent instrumentation, different reactivity mechanisms (rods vs. liquid poison injection), different manufacture of instrumentation where practical, different wiring, different trip parameters where practical, different design teams, different maintenance teams etc.
ASSESSMENT AND VERIFICATION OF SAFETY – COMPONENT ENGINEERING					
34	Germany		Introduction: p. 129	In Annex 1.1 describing the R & D Programs it is pointed out that "CANDU reactors were one of the first to make extensive use of digital control systems". As the application and the qualification of digital control systems and their software is a topic of general interest, what can be learned from experience made in Canada?	Use of digital control systems and lessons learned are discussed in Attachment 3.
35	Slovenia	14	Section 14.3.4: p. 84	… the feeders demonstrated leak-before-break behavior as expected. As the LBB methodology is not yet everywhere accepted by regulators, it would be interesting if you could explain this in more detail (comparison of feeder behaviour with predictions of LBB analysis…).	In more than 20 years of CANDU operation and 20,000 tubes in service, for the first time a feeder leaked at 600 MW Point Lepreau Power Station in late November 1996, channel S08, and a second one, channel K16, in late January 2001, at the same power station. Both are short radius outlet bends with a diameter of 2.5" (63 mm) and a wall thickness of 0.242" (6.15 mm) formed through cold bending. Laboratory investigations performed at AECL Chalk River showed

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					that the most plausible cause of failure is environmentally assisted cracking – Stress Corrosion Cracking (SCC) produced by high residual stresses from cold bending. The Leak–before-break assessment showed that: - there are adequate safety margins on applied load and crack size; - there are large margins between leak rate for which the unit is required by operational procedure to be shutdown and the leak rate that poses a threat to the structural integrity of a feeder. The estimated upper bound for the leak rate for which the unit has to be shutdown is around 500 kg/hr. This value is more than 15 times less than 8,500 kg/hr which is the predicted leakage when the axial crack reaches the "critical" crack length (90 mm) and more than 10 times the leakage at the time the unit was shutdown when S08 leaked; - it would take 20 days for a crack leaking at 30 kg/hr to grow to a size at which it would leak at 500 kg/hr. There is some physical evidence which confirmed the theoretical / leak-before-break predictions: - the expected maximum length of an axial crack that could develop due to the residual stresses in a 2.5" diameter small feeder bend, considering also the stiffening effect of the flange and the attached pipe, is 70 mm. At the time when the reactor was shutdown, S08 crack reached 63 mm inside (35 mm outside) and K16 crack reached respectively 55.1 mm inside (20.2 mm outside); - the predicted leak of a feeder of 2.5" (63 mm diameter) and uniform thickness of 0.242"(6.15 mm) is in the range of 40-60 Kg/hr. This is very close to the measured leak rate of 45 Kg/hr at the time of reactor shutdown in case of S08 failure.
36	Korea	14	Section 14.3.6: p. 86	It is described in 14.3.6 that "over the years, the environmental qualification process has not been well documented and there have been inconsistencies in the level of	The requirements for environmental qualification (EQ) originate from the Regulatory Documents R-7, R-8, and R-9. Collectively, they state that qualification is required for all equipment that is part of SDS, ECCS, and containment which may be required to operate or to

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				qualification provided." It is also stated that "licensees have been requested to develop the environmental qualification programs, implement the design and equipment changes, and develop programs to maintain the stations environmental qualification status." What are the regulatory requirements regarding the environmental qualification of safety-related components and materials and what kind of programs have been and will be developed by the licensees in terms of the environmental qualification?	continue operating following exposure to the harsh environmental conditions resulting from certain postulated events specified in these regulations. EQ programs developed by the utilities utilize industry standards and accepted methods to establish and preserve EQ. The technical basis for the program nominally is IEEE 323 standard for Qualifying Class 1E Equipment for Nuclear Generating stations, related daughter standards (addressing specific equipment) and draft CSA Standard N290.13, Requirements for EQ of Equipment for CANDU Nuclear Power Plants. The implementation model for EQ programs generally consists of the following phases: design basis or inputs, design verification, implementation and preservation. These elements are integrated into appropriate site processes and procedures to establish and maintain auditable proof of qualification through the life of the station. Completion of EQ Program is a condition in each plant's operating licence, as follows: "The licensee shall establish that all required systems, equipment, components, protective barriers and structures in the nuclear facility, are qualified to perform their safety functions under the environmental conditions defined by the nuclear facilities design basis accidents".
37	Korea		Annex 14.2: p. 227	It is addressed in Annex 14.2, "Required Improvements and Modifications for Restarting the Pickering "A" Reactors", replacement of emergency coolant injection shutdown cooling isolating valve actuators is one of the conditions that must be completed prior to restarting the reactors. Why is the replacement necessary and is it a unique problem to Pickering "A" reactors?	Operating experience at nuclear power plants in the 1980s and 1990s revealed a number of weaknesses which could adversely affect motor operated valve (MOV) performance. A common error in the initial design of MOVs resulted from inadequate prediction of the forces required to open and close valves under accident conditions. Both regulatory and industry research programs have since confirmed that the initial design calculations underestimated the dynamic loads experienced when the valves operated under flow conditions. In-service testing, consisting of measurement of valve stroke times under static conditions, cannot detect such deficiencies because they are conducted in the absence of dynamic loads. As a result, the USNRC

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					issued a Generic Letter (GL89-10) to ensure the capability of MOVs in safety-related systems to perform their intended functions at the conditions under which they must function. Licensees at Canadian plants have recently begun to address these MOV operability issues by implementing valve programs. In particular, OPG's Integrated Improvement Plan (IIP) included a comprehensive valve program, based on recognized industry guidance and best practice. The licensees at all Canadian NPPs have examined the operability of the emergency coolant injection valves, which must open against a high differential pressure during a large LOCA. Design basis review and valve operating margin calculations have revealed significant undersizing of the original valve actuators at the older plants. The marginal or insufficient capability of the ECI valves has been corrected at two plants by modifying the actuators. Actuator replacement was required both for EQ and to improve the design opening margin of the ECI valves at Pickering, so this modification was included in the restart requirements for Pickering A.
ASSESSMENT AND VERIFICATION OF SAFETY – AGEING MANAGEMENT					
38	Germany	14	Section 14.3.3: p. 82	Have the *"draft recommendations for a regulatory position on requirements for the management of ageing"* become an official regulatory document?	Ageing management is now included in the draft CNSC maintenance standard (C-210).
39	Japan	7&10	Section 7.4: p.39; Section 10.1.4 p.55	It is reported in section 7.4 that assessment of the licensee's safety performance is primarily through three kinds of activities, i.e. compliance verification activities, safety performance indicators and review of safety significant events, and the information is integral to the operating license renewal. Meanwhile, it is reported in section 10.1.4 that one of the mechanisms in accomplishing safety principles in the CNSC regulatory control is to establish a license renewal practice as a mechanism to ensure that there is compliance and periodic safety review.	Equipment fitness for service (including ageing) is discussed during licence renewal. This includes assessments of the licensees ageing programs as well as the results of compliance verification activities on the implementation of these programs. Licence conditions also require the results of equipment inspections done by the licensees to be submitted to the CNSC.

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				And is reported also that the license is typically issued for two years, but the Commission has the authority to issue shorter or longer term licenses. 1) In which mechanism in the operating license renewal process, is ageing effect on safety significant systems and components evaluated by utilities and by CNSC?	
40	Germany	19	Section 19.1: p. 116	For the Point Lepreau station a major systematic review of the special safety systems has been conducted which covered operating history and component ageing. Was the operating history recovered in such detail as to allow a comparison of observed ageing effects with analytical models? Are similar reviews planned for other stations?	The answer to the question is very dependent on the particular system and component involved. In the case of many of the special safety system functions, extensive commissioning tests and thorough data recording have allowed repeat performance tests to trend parameters and the plant to take corrective actions where appropriate. An example of this would be the Containment pressure testing which trended ageing effects on the epoxy containment liner and led to a liner replacement program. Not all operating history is recoverable in great detail from initial operation in 1982. Many parameters such as thickness profiles of feeder pipe elbows do not have 1982 baseline data, so ageing trends could only be based on more thorough data collection and trending initiated later in the plant life. Since 1992, system data for the Point Lepreau Generating Station has become more accessible to station staff. Prior to this, system data was only available through the Station Control computers, monitoring devices in the Main Control Room and various field indications. In 1992 hardware and software for extracting data from the Control Computers to an off line data computer was put in service (PLGS IR-05000-02, October 1994), which enables station staff to conveniently monitor system performance from their desktops. This has allowed retrieval and analysis of data from operational events, such as the observation of shutdown system performance, allowing trending at a level of detail that would otherwise only have been available through infrequent outage-related specific performance tests. Much more comprehensive System Health Monitoring programmes are now in place at Point Lepreau (SI-01365-T54) to allow observation of ageing effects.

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ASSESSMENT AND VERIFICATION OF SAFETY – SAFETY IMPROVEMENT PROGRAMS, PLANT RETURN-TO-SERVICE, PLANT REFURBISHMENT					
41	Germany	14	Section 14: p. 2, 77-90	What is the up-to-date requirement to be met by Pickering A with respect to an enhanced shut-down system?	The requirement for modern reactors is for two independent and diverse shutdown systems. Pickering A enhanced shutdown system does not fully meet this requirement. There is independence in detection of faults and trip initiation but both systems (SDSA and SDSE) drop the same shutoff rods. Other Canadian power reactors use poison injection for independence and diversity. The enhanced shutdown system (SDSE) covers all faults requiring rapid trip or overpressure protection; it does not provide duplicate trip coverage for all faults. Shutdown by moderator dump is available for slower faults. The CNSC accepted that the potential benefit over the remaining station life from making SDSA and SDSE completely independent was not justified by the cost in worker dose and additional monetary expenditure.
42	France	14	Section 14.3	OPG formed the Nuclear Performance Advisory Group to perform an Independent Integrated Performance Assessment. Could Canada clarify the composition of this Advisory Group?	The Nuclear Performance Advisory Group (NPAG) is no longer in existence. It was made up of seven US nuclear industry experts, who between them, had a total of approximately 200 years in the management, operations, support and assessment of nuclear power plants. Their collective experience was based primarily in the United States. Significant portions of the team's collective years of experience were spent at senior management and executive levels. The areas of expertise of team members spanned: plant start-up and recovery, operations, maintenance, design engineering, quality management and performance assessment. After assuming the management of OPG's (then Ontario Hydro's) Nuclear Division, team members assumed Vice-Presidential roles in each of the functional areas of the nuclear business. The NPAG group established assessment criteria as the basis for the execution of the Independent Integrated Performance Assessment (IIPA) to develop an integrated, accurate and comprehensive understanding of the weaknesses and strengths of the nuclear organization. The assessment criteria were derived from benchmarking US plants whose performance is ranked as excellent. The results of the IIPA

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					provided the basis on which a prioritized set of performance improvement programs were integrated into the Nuclear Asset Optimization Program (NAOP), which in turn formed the basis for improvement programs.
43	France	13	Section 13	It is indicated in the report that NPAG developed the Nuclear Asset Optimisation Plan to recover the performance of OPG's nuclear programme, which led to define the Integrated Improvement Program. One of the items of this programme is quality assurance and the CNSC review led to the introduction of a licence condition for this area. Could Canada give more information on the weaknesses of the quality assurance programmes, which were used on the nuclear plants, and what measures have been taken to improve them?	The essence of the Integrated Improvement Program was the development of an effective quality assurance program. OPG consolidated the numerous policies and procedures developed at the various sites into a single management program with standardized processes. The major weakness was effective implementation of the programs. Compliance was voluntary and determined by each site with no corporate oversight. Some of the weaknesses were 1) no one accountable for the business/management system; now there is a vice-president of managed systems and a Director of Performance Assurance to confirm implementation.2) No effective deficiency identification and corrective action program; now there is an effective reporting, analysis, corrective action and operating experience program. 3) No effective configuration management, now there is a strengthened engineering change control process.
44	Romania	10	N/A	The Report mentions that there are certain areas where the review acknowledged that Pickering "A" does not meet modern standards. These include shutdown systems, fire protection, main control room design, and seismic design. Which are the new licensing documentation which requires prior approval of CNSC and what specific actions regarding the preparation of the new licensing documentation are still outstanding and on what time scale will these be addressed ?	Question needs clarification. Details of the work required for the restart of Pickering A were presented in Annex 14.2.
45	Slovenia	14	Section 14.4: p. 87	Plant Return-to-Service and Plant Refurbishment Program. It is not clear whether a general quantitative safety related criteria has been established already before the detailed analysis of return-to-service NPP's has been performed. Could you please elaborate on quantitative general safety	A quantitative general safety criterion has not been established for return-to-service plants. A comprehensive review is completed against current requirements and agreement is reached between the regulator and the licensee on the requirements for return-to-service.

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				criteria (regulatory requirements) that return-to-service plants have to fulfill.	
QUALITY ASSURANCE					
46	France	13	Section 13.1	It is indicated in the report that the nuclear regulator can impose additional requirements to the standards developed by the Canadian Standards Association. Does Canada plan to produce a regulatory guide, which covers all the quality requirements of the nuclear regulator?	The CNSC has no current plan to produce a regulatory guide. The Canadian Standards Association develops consensus standards through committees composed of industry and regulator representatives. The CSA standards for procurement, design, operations, etc., have recently been consolidated into a single standard which will be supported by a Guide. The CNSC will evaluate the completed guide for adequacy.
47	USA	13	Section 13.2: p. 74	The report indicates that quality assurance standards define what safety-related means and that the regulatory body requires licensees to identify safety-related items, activities and processes in accord with the definition. What has been your experience in grading quality assurance requirements according to safety significance?	The grading of quality assurance requirements according to safety significance was found to be quite effective in manufacturing. A contractor could be awarded a CSA Z 299.1 (highest level) contract for design, assembly, and testing of a pump and subsequently assign different quality program levels to subcontractors (e.g., CSA Z299.3 for a pump motor and CSA Z299.2 for pump pressure boundary components, with commercial grade for components with no significant safety function). This approach was not entirely successful with operating nuclear power plants. Operators identified safety systems and safety support systems as required and ensured the appropriate quality assurance was applied but this approach is not as effective as having a single integrated management system. The evolution of quality assurance into quality management has resulted in a broader approach. An integrated program applicable to all systems and components provides superior results. The CNSC currently reviews the overall management system to ensure that an operator has an effective process for identifying and applying the appropriate quality assurance requirements for all components. The operator, for example, may decide to apply quality assurance requirements to a component to safeguard against financial risk. In summary the CNSC expects the quality assurance requirements to be determined on a case by case basis as part of an integrated process rather than only being applied to items on a particular list.

	Country	CNS Article	Report Reference	Question	Answer
48	France	13	Section 13.3	Is there a system to collect and analyse the events related to quality in order to implement a continuous improvement of the quality assurance system?	Yes, for example, OPG has established a corrective action and operating experience program that captures even minor events, categorizes them, trends them, assigns a root cause and has a corrective action program that results in continuous improvement.
49	Germany	14	Section 14.2: p. 78-79	Have the self-assessment programmes been in place since operation began at the various plants or were they introduced at a later stage?	These programs were not in place at the start of plant operations. For example, at Point Lepreau operations started in 1982. Self-assessment was undertaken informally by some workgroups in the 1980's and early 1990's, however their impact was minor due to the very high ranking production success of the station from 1983 to 1994. Following a series of incidents during the 1995 outage it became evident that there were significant problems with many of the programs at Point Lepreau and NB Power undertook an extensive review of our approach to Nuclear Management (AECB BMD 97-54, April 1997). This was essentially a comprehensive Management self-assessment aimed at Improving Safety Performance at Point Lepreau. A comprehensive station-wide formal self-assessment programme, following WANO recommendations, was introduced during 1998 and early 1999 (PLGS SI-01365-A62, January 1999).
HUMAN FACTORS					
50	Slovenia	12	Section 12.1.1: p. 66	"Canadian nuclear organization have active research and development programs to support both short and longer term design, operations and regulatory needs. Seminars, reports and conference papers are used to disseminate the results. Recent topic areas include: development of a systematic method for regulatory assessment of licensees' organization and management." Could you please elaborate a little bit more on this interesting topic such as objectives, scope, main elements of licensee organization and management assessment, etc.?	A full description of the process is available in AECB Report RSP 0060 "Development of a Regulatory Organizational and Management Review Method" which is available from the CNSC Library. A briefing note on this subject is included as Attachment 4.
51	Slovenia	12	Section 12.3.1:	"CNSC activities in the area of human factors include: • review of significant design	There is currently no formal CNSC approval. However, the CNSC has issued the regulatory document "P119: Policy on Human

	Country	CNS Article	Report Reference	Question	Answer
			p. 70	modifications and organizational changes; • audits and evaluations of licensee programs which impact on human performance (e.g., corrective action/operating experience, engineering change control); and • development of human factors regulatory documents". Do these activities include also the CNSC approval ? Could you explain what criteria and methods/procedures for assessment are used?	Factors". It has also issued documents "C276: A guide to Human Factors Engineering Programme Plans" and "C278: A guide to Verification and Validation" for consultation.
52	Korea	12	Section 12.3.2: p. 71	What are the evaluation items and operational considerations of the Canadian Adaptive Machine Model (CAMM) determining the integrity of safety culture?	Refer to response to Q50.
53	Germany	18	Section 18.1: p. 111, 245	Are installations like full-scope simulators and virtual reality tools being used or considered for training? Is there a systematic review procedure in place to identify human factor influences when analysing abnormal events?	Full-scale simulators are used by Canadian NPPs for training. Virtual reality tools are not in use. All NPP licensees have systematic root cause analysis processes in-place which include the identification of Human Factors influences.
54	France	12	Section 12	Could Canada provide some information about the emergency operating procedures used by the different plants? Are they event-oriented or symptom oriented? Are the PSA results used for the definition of the operators training (identification of critical actions)?	The emergency operating procedures at the different plants are mostly event-based. A limited set of symptom-based procedures have been introduced at some plants. PSA results are accounted for in these procedures.
55	France	10	Section 10.1-10.2	Could Canada indicate if external reviews have been conducted by international organisations for assessing safety culture?	External reviews have been conducted by WANO at nuclear power plants in Canada. These reviews consider elements of safety culture.
56	Korea	12	Section 12.1.1: p. 66	What is the operating system of the human factors "champions" in the AECL and what are the effects?	We believe the question pertains to AECL's Branch HF Champions. This designation was used only on CANDU 9 and reflects AECL's most current implementation of HF in design. The Branch Champions (BCs) were designers in each of the major design areas (e.g., process, civil, I&C) who were given HF training to the level where they could assist other designers in their branch to identify and resolve many design problems related to HF. The BC would also quickly identify the need for HF specialist support. Thus the role of the BC was to

	Country	CNS Article	Report Reference	Question	Answer
					respond knowledgeably to emerging HF design issues across the CANDU 9 project on a day-to-day basis. It is not possible for HF specialists to provide this level of coverage and support across the entire project. Project specific HF Design Guides were developed for the project in support of BC work. The overall integration of HF into design is governed by a project specific Human Factors Engineering Program Plan which maps out an integrated approach across the design. The programmatic level guidance for the HF program comes from IEEE 1023 and NUREG 0711.
57	Romania	14	N/A	How is guaranteed that the staff who monitor safety are not influenced by production needs?	To ensure that staff who monitor safety are not influenced by production needs, it is necessary for the Management System to establish and maintain a Safety Culture throughout the station which puts safety first, recognizing and taking seriously the unique safety requirements of the nuclear core. Having established such values, it is important to ensure that actions of the management team are seen by staff to re-enforce, not contradict, this value system. Further, it is necessary to ensure that the reward system for staff who monitor safety (which should be ALL station staff), is such that the right attitudes, characteristics and actions towards safety are rewarded and that any attitudes, characteristics and actions which tend to favour production over safety are not rewarded. The reward system not only involves money. To the extent that money is offered as a reward, it must be focused on the cultural goals. For example, the 2000 Outage at Point Lepreau was targeted to take 77 days. All PLGS staff except management were offered a bonus to complete the outage up to three days earlier. In order to offer the bonus in the right cultural context, the bonus was offered for payment 30 days after the completion of the outage, but only if the following conditions were met: • *Personal Safety*: Two or fewer lost-time accidents during the outage period (lost time defined as not reporting to work for the next scheduled shift) • *Nuclear Safety*: Two or fewer International Nuclear Event Scale level 1 events during the outage period.

	Country	CNS Article	Report Reference	Question	Answer
					• *Quality*: The unit must run continuously for 30 days without breakdown or trip after the outage. Point Lepreau view this bonus incentive as having been successful, and the extra three days of production were achieved with greater overall safety, through greater team work and co-operation between station work groups, all of whom were rewarded for helping each other achieve their goals more safely. A CNSC staff Human Performance evaluation of the incentive scheme did not contradict this Point Lepreau perspective.
STAFFING AND TRAINING					
58	Germany	19	Section 19.2: p. 119-121	The operating licence also specifies the minimum staff compliment. Is the minimum staff compliment different for various plants and what are the determining factors? Has it been modified in licence renewal procedures?	The minimum staff complement is similar for plants with the same number of reactors, i.e., four-reactor plants or single-reactor plants. Except for a few positions covered specifically in the licences, the minimum complement is specified in plant administrative documents, which receive regulatory approval. The minimum complement is the minimum number of people required to ensure safe and reliable operation of the plant, while maintaining an adequate preparedness level for responding to all emergency scenarios. This number must be sufficient to handle operations of the plant, necessary responses in the plant, the exclusion area and off-site, and station security. The minimum complements have not changed significantly over the years.
59	Korea	11	Section 11.5: p. 60	Is there regulatory requirement for surplus personnel for training to assure safe operation? If yes, what is the size of surplus personnel?	The question is interpreted as follows: Is there a regulatory requirement for each plant to have personnel, in addition to the number necessary to perform the needed work, who would be in training and preparing to replace qualified persons who may leave? There is no such requirement for anticipatory training.
OPERATIONS AND EVENT ANALYSIS					
60	Korea	19	Section 19.2.7: p. 120	It is stated that there are currently 18,000 records in the database of CNSC. Is it possible to access from the outside? If it is	No, not at the present time. Access may be made available in the future on a bilateral basis.

	Country	CNS Article	Report Reference	Question	Answer
				possible, how can the information be obtained?	
61	Korea	19	Section 19.2.7: p. 120	It is stated that the Regulatory Event Assessment Program (REAP) was set up by the CNSC in 1998. Are there any regulation-based obligatory feedback items? If so, what are the items?	The CSA standard on Quality Assurance requires the licensees to have programs on Operating Experience. This CSA standard is a condition of the operating licence for the power reactor.
62	Germany	19	Section 19.2.7: p. 119	Does the Regulatory Event Assessment Program (REAP) include a precursor analysis?	No.
63	Germany	6	Table 6.8: p. 25	In judging the influence of the Point Lepreau performance improvement programme (PIP), performance indicators different from those used for OPG were listed. Is it intended to define a set of performance indicators to be used by all plants? Has the accident severity rate,, which for Point Lepreau "dropped significantly" from 1996 to 2000, been measured for other plants as well?	CNSC introduced a common set of performance indicators for all licensees in 1999. Accident severity rate is one of those measures. The CNSC does not use performance indicators in isolation but in combination with event and assessment reviews. The accident severity rate performance indicator has been used to direct some resources for further investigation at different plants.
64	Germany; Slovenia	6	Table 6.4: p. 22	Have reasons been identified for the consistently high number of "Reportable Events" at Darlington compared to the other plants?	The vast majority of the reportable events at Darlington are related to impairments of steam protection doors. Doors, seals, steam traps, dampers, ventilation and air conditioning systems form an integral protective physical barrier to protect sensitive safety related equipment from the potential harsh environments which may arise from design basis accidents such as a break in a steam or feedwater line. The defence-in-depth philosophy employed at Darlington is such that any impairment to the protective envelope, such as improper closure or sealing of a steam protection door, is regarded as an impairment to the system, and reportable to the regulator. The vast majority of these events occurred as a result of improper closure of the steam doors by workers upon entering or exiting rooms which are equipped with these doors. With regard to the reported event statistics, the number of reportable Darlington events would compare favourably with those of OPG's other plants, should the steam door category be deleted. Station management has taken several steps to address the problem. A station responsible engineer was assigned for the control of steam

	Country	CNS Article	Report Reference	Question	Answer
					barrier design and maintenance of barrier integrity. Closure mechanisms for most doors have been replaced and station staff coached on the importance of maintaining the integrity of the steam protected envelope. Inspections and testing are carried out regularly. As a result of these initiatives the number of reportable steam-protection door impairments was reduced from 29 in 2000 to 6 in 2001.
65	Slovenia	6	Section 6.5: p. 22	Table 6.4, Reportable events in 2000: for Darlington 3,5; Bruce 3; Pickering 4. What is the nature of reportable events at Darlington NPP?	As per our response to the Q64, the majority of reportable events for Darlington arose from improper closure of steam doors, resulting in impairments to the plant's protective envelope. In year 2000, 29 steam door events were reported for Darlington. This number was reduced to 6 in 2001 as a result of corrective actions described in the previous answer. The breakdown of the 35 events at Darlington in 2000 is as follows: 29 events were related to steam door impairments. 1 event was a unit 4 SDS1/SDS2 trip. 1 event was interference with boiler SRV manual latches with the actuator external ring allen screws. 1 event was unit 2-67210-TCV14-2 SV tubing installed incorrectly. 1 event was the LISS helium supply check valve possibly not being reliable. 2 separate events were holes through beams in a seismic stairwell.
66	Japan	6 & 9	Section 8: p.8; Section 19.3: p.98	Table 6-6 shows the capacity factor from 1998 to 2000. How many days is the average outage duration per unit? Is plant outage for maintenance a regulatory demand? If so, how often?	Refer to Attachment 5 for information on the outage duration per unit. There are no specific regulatory requirements on the management of and frequency of plant outages. The CNSC requires that the licensee, through the operating licence, maintain the plant in a condition such that the reliability, and effectiveness of any structure, system, or component remains consistent with the description and analysis provided in the Safety Report. This is achieved through the plant routine maintenance program,

	Country	CNS Article	Report Reference	Question	Answer
					which includes testing and inspection. Some maintenance work on safety related systems is carried out during outages. The outages are managed and executed in accordance with procedures to ensure that adequate safety margins are maintained during the conduct of outage maintenance work. Extensive planning and procedural application are only required to ensure effective and efficient outage execution. The CNSC monitors and assesses the licensee's management of maintenance outages as part of its licensing performance evaluation. Specifically, the CNSC evaluates dose expenditures and the effectiveness of reactor safety management during the outages.
EMERGENCY PREPAREDNESS AND RADIATION PROTECTION					
67	Germany	18	Section 18: p. 111, 242	Among the barriers to radioactive releases, the exclusion zone is to provide atmospheric dilution of any fission product releases from the containment. What is the typical size of an "exclusion zone"? Do the safety analyses take credit of these zones? Are weather conditions taken into account in the analyses?	The exclusion zone in Canada has traditionally been land within 1000 yards (914m) of the containment. No permanent dwelling may be built inside the exclusion zone. Safety analysis calculates the more limiting individual doses at the site boundary (not the exclusion zone boundary). The public dose is based on the actual population distribution outside the exclusion boundary. All analyses assume adverse weather conditions (Pasquill F).
68	Germany	15	Section 15.2: p. 92, 233-235	Does the indicated dose rate limit of 1 mSv/y refer to the population in the vicinity of a plant, to adults or to infants? Would this limit ensue as a total if the Derived Release Limits of all nuclide groups mentioned were applied?	For members of the public, the effective dose limit is an annual effective dose of 1 mSv, and is not specific to an adult or infant. This limit is not related to an emergency situation, but rather is the limit for a person who is not a nuclear energy worker. The limits for nuclear energy workers are 50 mSv over a 1 year dosimetry period in this case and 100 mSv over a 5 year dosimetry period. The DRL for each type of release (e.g., HTO, noble gases, iodine or any other) is set so that the resulting dose to the most exposed member of the public does not exceed 1mSv (1DRL ↔ 1 mSv). Since stations run well under the DRLs, typically 5% of the DRL for each radionuclide, a release of a mix of radionuclides does not result in exceeding the 1 mSv dose associated with the DRL. During the control of an emergency and the consequent immediate

	Country	CNS Article	Report Reference	Question	Answer
					and urgent remedial work, the effective dose, and the equivalent dose may exceed the regulatory limits for the emergency workers, but the effective dose shall not exceed 500 mSv and the equivalent dose received by the skin shall not exceed 5000 mSV.
69	Germany	15	Sections 15.3, 15.4: p. 92-93	Is there a control mechanism (independent laboratory measurements) for the measuring results from the emission measurements carried out by the NPP-operating utilities? In which way is quality control executed in connection with emission and immission measurements?	There are no independent release measurements conducted by the CNSC or any other regulatory agencies. However, quality control of release monitoring is done through evaluations of licensees environmental protection programs by the CNSC. In addition, there is an inter-utility instrument calibration service in place.
70	France	16	Section 16.1	In the report, 3 on-site and off-site emergency plans are presented and the descriptions are quite different. a. Are there guidelines, produced by CNSC or derived from international documents, used to develop these plans?	There is a draft regulatory guideline document that provides evaluation criteria for on-site emergency response plans and programs. The purpose of the proposed guidelines is to establish a baseline against which off-site nuclear emergency planning and response can be measured and to provide a generic basis for the harmonization of provincial plans. Other Canadian guidelines such as the one for the restriction of radioactively contaminated food and water following a nuclear emergency, or the draft document for intervention following a nuclear emergency are published by Health Canada.
71	France	16	Section 16.1	In the report, 3 on-site and off-site emergency plans are presented and the descriptions are quite different. b. Could Canada give more information on the criteria used to enter in an emergency situation?	In Canada, there is no commonly agreed-upon system to unequivocally categorize and communicate accident severity among intervening organizations. Typically, the accident classification scheme takes into account the nuclear facility conditions, the safety systems status, the potential of environmental releases, the measured environmental releases and the result of radiation monitoring. However, within a province on-site and off-site authorities use the same classification system which triggers automatic notification or, in some cases precautionary protective actions. The federal, provincial and local emergency measures organizations are responsible in varying degrees for the protection of the public. The criteria used to define different actions to protect the public vary from one province to the next. In an effort to harmonize emergency responses, Heath Canada has drafted guidelines for intervention following a nuclear emergency where:

	Country	CNS Article	Report Reference	Question	Answer
					COUNTERMEASURE and Intervention Level (averted dose) Evacuation: 50 mSv in 7 days Relocation: 50 mSv in 1 year; return when ≤ 10 mSv in 1 month Sheltering: 5 mSv in 1 day Stable Iodine Prophylaxis: 100 mSv to thyroid Food Controls: 1 mSv from each of 3 food groups
72	Germany	16 (1)	Section 16.1.2: p. 97	Which emergency reference levels are defined for the implementation of the emergency protective actions?	Refer to answer for Q71
73	France	16	Section 16	In the report, 3 on-site and off-site emergency plans are presented and the descriptions are quite different. f. Is the public living in the vicinity of the plants aware of the protection actions of the public in case of an emergency?	One of the off-site emergency plan requirements for the nuclear facilities is a Public Education Program to assure public understanding of how to participate and cooperate effectively in the event of an emergency.
74	Slovenia	16	Section 16.1.2: p. 97	What is the strategy for delivering iodine tablets?	The strategy varies with each province to address their specific needs. The province of New Brunswick has arranged for prior distribution of stable iodine because there are only a few hundred scattered residents in the vicinity of the Point Lepreau NGS. The province of Quebec contingency plans require prior distribution of stable iodine to emergency workers and on-site personnel of the Gentilly Industrial park and the Bécancour shipyard. There is no prior distribution of iodine tablets in Ontario.
75	Germany	16 (1)	Section 16.2.2: p. 99	Are emergency plans on the regional level harmonised regarding content or structure in Canada? Are there any distances defined to which counter measures should be considered or pre-planned ?	Refer to answers for Q67 and Q71
76	France	16	Section 16	In the report, 3 on-site and off-site emergency plans are presented and the descriptions are quite different. d. Are there computerised support systems to predict the accident progression and to predict the doses around the plant?	Yes, several nuclear facilities have plume dispersion models, weather data accessibility and make use of off-site survey teams to follow the accident progression and assess doses around the plant.
77	France	16	Section 16	In the report, 3 on-site and off-site emergency	Refer to answer for Q71

	Country	CNS Article	Report Reference	Question	Answer
				plans are presented and the descriptions are quite different. g. What are the criteria to define the different actions to protect the public?	
78	France	16	Section 16	In the report, 3 on-site and off-site emergency plans are presented and the descriptions are quite different. h. Is iodine distribution a protection measure used?	Refer to answer for Q74.
79	France	16	Section 16	In the report, 3 on-site and off-site emergency plans are presented and the descriptions are quite different. e. In the off-site emergency plans, is there an emergency building where all information concerning the accident is collected?	All organizations involved in the response and in the mitigation of a nuclear emergency have their own Emergency Operating Centres (EOC) at the facility, municipal, provincial, and federal level. There is no automatic data transmission to these centres at this time.
80	France	16	Section 16	In the report, 3 on-site and off-site emergency plans are presented and the descriptions are quite different. c. Are the nuclear plants equipped with an emergency building?	Refer to answer for Q79

ATTACHMENT 1 Additional Information for Q5

Historically in Canada, licences for nuclear facilities have been issued for a period of two years. This has permitted close scrutiny of the licensee by the Commission and offered frequent opportunities for public intervention. Some variations to the standard two-year term have been granted, ranging from only a few months to indefinite periods. Reduced licence terms have been associated with poor performance and longer licence terms have usually been granted on the basis of risk of the facility.

These relatively short licence periods have permitted close scrutiny of the licensee by the Commission and offered frequent opportunities for public information and intervention. This open and transparent regulatory approach provides Canadians with assurance that the regulated facilities are operated safely.

However, a two-year licence period is often not long enough to enable staff to complete a full review of the licensee's activities, and the heavy resource requirements of licensing require the redirection of staff away from inspections, compliance-verification activities, and performance audits. This short licence cycle has also resulted in licensee resources being diverted from activities that might have a greater impact on assuring safety of the facility. It has become apparent that a longer licence period would permit more in-depth compliance activities over the licence period thus permitting staff to present the Commission with a more comprehensive evaluation of licensee performance and facility safety when considering licence renewal. In particular, adoption of longer licence periods for power reactors with periodic safety reviews during the licensed period would align the CNSC's regulatory regime more closely with those of other countries.

As such, Canada's traditional approach with a relatively inflexible two-year licence period is not consistent with risk-based regulation and does not permit the most effective use of regulatory effort. CNSC staff has, therefore, proposed a set of criteria that provides a systematic basis for recommending licence periods that will reduce unnecessary regulatory burden while maintaining a rigorous level of control.

The criteria include:

- The recommended duration of the licence should be commensurate with the licensed activity.
- A longer licence period can be recommended when the hazards associated with the licensed activity are well characterized and their impacts well predicted, and they are within the scope considered in the environmental safety case.
- A longer licence period can be recommended when licensees have in place a management system, such as a quality assurance program, to provide assurance that their safety-related activities are effective and maintained.
- A longer licence period can be recommended when effective compliance programs are in place on the part of both the applicant/licensee and the CNSC.
- A longer licence period can be recommended when the licensee has shown a consistent and good history of operating experience and compliance in carrying out the licensed activity.
- The licence period should take account of the planning cycle of the facility, and the licensee's plans for any significant change in the licensed activity.

As has been the case in the past, a recommendation by staff for a licence period of two years or less will continue to be an option where overall licensee performance is unsatisfactory.

In order that the Commission and the Canadian public are fully aware of current licensee performance, regular public reporting on licensee performance would replace 2-year licence renewal, thus ensuring that the nuclear regulatory regime remains open, transparent and accountable to the public. Furthermore, to maintain a high degree of transparency, regular compliance reports will be made to the Commission and the Canadian public during the period of each licence.

ATTACHMENT 2 Additional Information for Q13

On April 25, 2001, the federal government tabled Bill C-27, the proposed *Nuclear Fuel Waste Act*, to address the long-term management of nuclear fuel waste in Canada. Bill C-27 would require the Canadian nuclear utilities (Ontario Power Generation Inc., Hydro-Québec, and New Brunswick Power Corporation) to establish a waste management organization (WMO) as a separate legal entity. The purpose of the WMO would be to:

- propose to the Government of Canada approaches for the management of nuclear fuel waste; and
- implement the approach selected by the Governor in Council.

Bill C-27 would require that, within three years of the coming into force of the Act, the WMO submit to the Government an options study setting out its proposed approaches for the management of nuclear fuel waste, and its recommendation on which approach should be adopted. The bill would require the WMO to carry out public consultations as part of the options study. Bill C-27 specifies that the study must include the following approaches:

- deep geological disposal in the Canadian Shield;
- storage at nuclear reactor sites; and
- centralised storage, either above or below ground.

Bill C-27 would require the Governor in Council, on the recommendation of the Minister of Natural Resources, to select one of the approaches for the management of nuclear fuel waste from among those set out in the options study. The WMO would then be required to implement the selected approach.

Bill C-27 would require the nuclear utilities and Atomic Energy of Canada Limited (AECL) to establish trust funds, and to make annual payments into those trust funds, to finance long-term nuclear fuel waste management activities. As such, the nuclear utilities and AECL will continue to be financially responsible for the management of the waste. Bill C-27 specifies that the WMO may only make withdrawals from the trust funds for the purpose of implementing the approach selected by the Governor in Council.

ATTACHMENT 3 Additional Information for Q34

Digital Control Systems in CANDU Reactors

The rationale to use computers in the shutdown systems included equipment cost savings, better space utilization, capability to use complex trips (setpoints that are a function of power), reduced operator load for testing and calibration, and increased safety reliability achieved by early fault detection through monitoring functions.

The normal control of the major functions in CANDU (e.g., reactivity control, boiler pressure control) is via two redundant digital control computers. They have proven highly reliable in service, such that dual computer failure is not a major contributor to station unavailability. Aspects of the design which achieve high reliability are:

- redundant computer control, with one computer as 'master' and the other continuously running in parallel processing the same inputs and outputting (unused) signals;
- continuous internal self-checking, so that a detected programme fault results in transfer of the affected programme to the standby computer;
- a watchdog timer, such that if a computer stalls, overall control is transferred to the standby computer; and
- separate setback and stepback routines, less dependent on the main control loops, which act on abnormal indications and reduce the reactor power or shut it down.

From a safety point of view, an a dual computer stall results in the reactor shutting down through the control devices; this is of course backed up by the two shutdown systems. In recent designs, the display function has been separated from the control function. Separate computers are also used to control on-line refuelling.

Monitor computers were installed in the Bruce "A" and "B" units to upgrade the operator interface. These computers provide a bar chart displays for a CRT display. They provide warning to the operator if it detects variables too close to the setpoints, failed signals or disagreement among signals in the three channels. Trip Computers, Programmable Digital Comparators, replaced the function of the analog comparators and the associated conditionings in the CANDU 6 plants. The following advances were made for the Darlington units. Display computers replaced the conventional panel meters in CANDU 6 plants. Test controls were replaced by Test computers through which preprogrammed tests are initiated. Monitor computer made the manual amplifier gain adjustments be replaced by software gain factors down-loaded from the monitor computer. Trip computers perform all trip logic and conditioning (which includes the Bruce and CANDU 6 computer features). The trip computers also eliminated the inconveniences experienced at Bruce related to testing and calibration.

Lessons learned from CANDU experience are summarized below.

1 Operating Interface Requirements
   - Operator interface should be consistent with the remainder of the control centre, i.e., based on CRTs and keyboards.
   - System must warn the operator to take appropriate action for the off-normal conditions.
   - Tests should be automated to relieve the operator of repetitive tasks.

2. Separation Requirements
   - Different hardware and software should be used in the two shutdown systems to prevent any possibility of common mode failures, if feasible.
   - The links between the monitor and the channelized display / test computers must have electrical isolation, e.g., optic coupling.
   - The links must also be functionally buffered, to prevent a failure in the monitoring computer from affecting all three channels.

3. Performance Requirements
   - Trip computers must act quickly in less than about 100 ms to handle the most severe accidents
   - Channelized displays of trip signals, setpoints must be updated at approximately one second intervals.
   - Monitor computers must store approximately 12 hours of historical data.
   - Trip computers must be capable of performing local coincidence trip logic voting.

4. Reliability Requirements
   - Trip computers must meet the reliability requirements as follows:
   - shutdown systems must be unavailable less than 10^{-3} of the time, and this performance must be confirmed by regular testing. The target is to have fewer than 0.1 spurious trips per year in each shutdown system.
   - Trip computer hardware and software must be kept as simple as possible to maximize reliability.
   - Trip computer must fail safe, if possible, and should contain comprehensive self checks to ensure that all important components are operating correctly.

5. Software Verification/Validation Requirements
   - Validation tests are planned and executed by staff independent of the trip computer programmers.
   - Software engineering process includes Design Input Documentation, Requirements Definition, Software Requirements Specification, Design, Software Design Description, Code Implementation, Source Code, Executable Code, Databases.
   - It also includes Hazards Analysis, Reliability Qualification.

ATTACHMENT 4 Additional Information for Q50

REGULATION OF ORGANISATION AND MANAGEMENT OF CANADIAN NUCLEAR FACILITIES

At the Canadian Nuclear Safety Commission, we have developed a method and process for assessing the effects of Organization and Management (O&M) influences on nuclear safety. In the longer term our goal is to use these tools in order to predict when safety in a nuclear facility is likely to decline. That is an ambitious task, given the magnitude of the data required to do those types of predictive analyses. We are not there yet, however, we are in the midst of doing just that. We are developing an overall plan to examine what steps need to be taken to develop a set of predictors that can be used confidently and reliably.

The first phase of our work to achieve the above goals was to develop an assessment method that can be applied to Canadian nuclear facilities. The method, known as the Organization and Management Review Methodology [1], includes a model of the organization, called the Canadian Adaptive Machine Model (CAMM), and examines the human organizational characteristics that influence safety in a Canadian nuclear facility. The model, based on the work of Henry Mintzberg [2], postulates that a nuclear organization can be configured into five components that include the Strategic Apex (to set the corporate vision, goals, and policies and translate them into site goals, and policies), a Middle Line (to oversee activities related to operations, maintenance and service), a Technostructure (to standardize work processes, outputs and the skills of the operating professionals), an Operating Core (to accomplish the work of the organization) and Support Staff (to facilitate work and minimize any disruptions to the flow of work). Hypotheses related to the organizational and management functions, and processes related to safety can then be generated and measured.

Organizational factors (management oversight, organizational clarity, communication, organizational culture, and human resource management), which contain nineteen dimensions are attributes that influence the organization and can be measured using both qualitative and quantitative measures. First, there is a functional analysis of the organization, followed by structured interviews, an organization culture inventory, behavioral observations and behavioral anchored rating scales. The results of these assessments provide a descriptive profile of the facility, showing those processes that are working well and where improvements are needed.

Having developed and validated the assessment method, we have completed the O&M evaluations of all nuclear power plants in Canada and a number of other nuclear facilities such as research reactors, uranium mines and mills, accelerators, etc. Data are beginning to emerge that confirm that nuclear facilities in Canada belong to a small population of organizations known as "high reliability organizations", as defined in the literature [3]. Behaviors that would be exhibited in high reliability organizations have been grouped together in what has been termed a Constructive-Affiliative cultural style. This consists of constructive values, a drive to perfection, commitment to the organization, work group cohesion, work coordination, job satisfaction, open and effective communication and an emphasis on safety. Those kinds of organizations tend to be perfection-seeking, with an orientation towards a safety culture, more so than non-nuclear organizations. Conversely, the existence of most characteristics of Passive-Defensive (characterised by the descriptors Approval, Conventional, Dependent and Avoidance) and Aggressive-Defensive (Oppositional, Power, Competitive and Perfectionistic) cultures, and their corresponding behaviors, are expected to be absent (or very low) in high reliability organizations. All Canadian nuclear facilities evaluated to date exhibit high reliability characteristics.

Although the CNSC is not yet at the point where, based on these few O&M evaluations alone, it can decide that action needs to be taken to stop a decline in safety standards, the use of all of its O&M evaluation data, combined with other information provided by inspections and audits, provides the CNSC with a profile of the organization that it *can* and *does* use in its oversight and boundary monitoring of its nuclear licensees. The high reliability characteristics exhibited by the nuclear facilities evaluated to date has been factored into our regulatory decisions. That is not to say that there is not room for further development. We have, however, taken the view that if other key results areas are acceptable, then we can conclude that the organization and management profiles are likely to be acceptable as well. The other key areas are defined in the eighteen Technical Programs of the CNSC's Compliance Program and include, for example, Criticality

Safety, Emergency Preparedness, Radiation Protection, Human Factors, Environmental Protection, Fire Protection, Quality Management, and Training Program Evaluation.

By the end of 2001, baseline assessments had been completed for most of the major licensees that the CNSC regulates. Ongoing monitoring of all of those licensees will continue into the future, as part of the Compliance Program. Subsequently it is our intention to periodically revisit and re-evaluate those sites in order to monitor them for any changes to their profiles. It is important to note that organizational change can occur insidiously over a long period of time and through informal processes, as well as through planned and managed change initiatives. In our view, therefore, it is imperative that the regulator keep a watching brief on O&M issues on a continuous basis as part of the normal regulatory overview. If a nuclear facility's O&M profile changes away from the theoretical characteristics of a high reliability organization, resulting in a potential narrowing of the safety margin, other technical information will then be examined to provide some insight into the causes of those changes. Regulatory action will then be taken based on the objective analysis of all of those data.

By identifying and correlating O&M performance indicators from those evaluations with other existing/developing performance indicators, it is our intention that, in the future, those data will be predictive of situations that are harder to discriminate in terms of safety performance.

Reference

[1] Haber, S.B. and Barrier, M.T. (1998). Development of a Regulatory Organization and Management Review Method. AECB Research Report RSP-0060, CNSC, Ottawa.

[2] Mintzberg, H.T. (1983). Structure in Fives: Designing Effective Organizations. New Jersey:Prentice-Hall.

[3] Haber, S.B. and Shurberg, D. A. (1996) . Safety Culture in the Nuclear Versus Non-Nuclear Organization. Proceedings of the 1996 Probabilistic Safety Assessment Meeting, Seattle, WA.

ATTACHMENT 5 Additional Information for Q66

The table below shows the number of outages and the average outage duration per unit in days from 1998 to 2001 for Ontario Power Generation (OPG) reactor units.

Average Outage Duration Per Unit (days)

Unit	**1998**		**1999**		**2000**		**2001**	
	No. of Outages	Average Duration	No. of Outages	Average Duration	No. of Outages	Average Duration	No. of Outages	Average Duration
Bruce Unit 1	-	-	-	-	-	-	-	-
Bruce Unit 2	-	-	-	-	-	-	-	-
Bruce Unit 3	1	1.9[(1)]	-	-	-	-	-	-
Bruce Unit 4	2	36.6	-	-	-	-	-	-
Bruce Unit 5	6	10.1	1	85.0	2	1.4[(2)]	0	0
Bruce Unit 6	3	36.5	0	0	2	59.8	1	2.0 [(3)]
Bruce Unit 7	7	13.6	3	7.7	2	39.2	0	0
Bruce Unit 8	2	70.7	7	19.9	2	10.2	1	66.3
Darlington Unit 1	4	10.9	1	2.3[(4)]	1	48.7	2	5.4
Darlington Unit 2	3	17.6	2	19.5	3	7.8	4	18.0
Darlington Unit 3	2	3.5	2	38.2	4	10.0	5	7.2
Darlington Unit 4	1	44.2	4	13.9	2	11.8	1	30.1
Pickering Unit 5	3	20.3	2	72.0	4	34.7	3	38.5
Pickering Unit 6	2	49.5	2	39.5	3	32.4	3	48.3
Pickering Unit 7	3	31.5	0	0	2	90.4	6	5.5
Pickering Unit 8	5	14.6	2	35.1	2	68.3	3	24.5

Notes: Data for Bruce available only up to end of May 2001.
(1) Forced outage due to failed controller; returned to service after poison out period.
(2) Forced outage due to boiler level control problem.
(3) Forced outage due to computer controller fault.
(4) Forced outages due to failed control absorbers.