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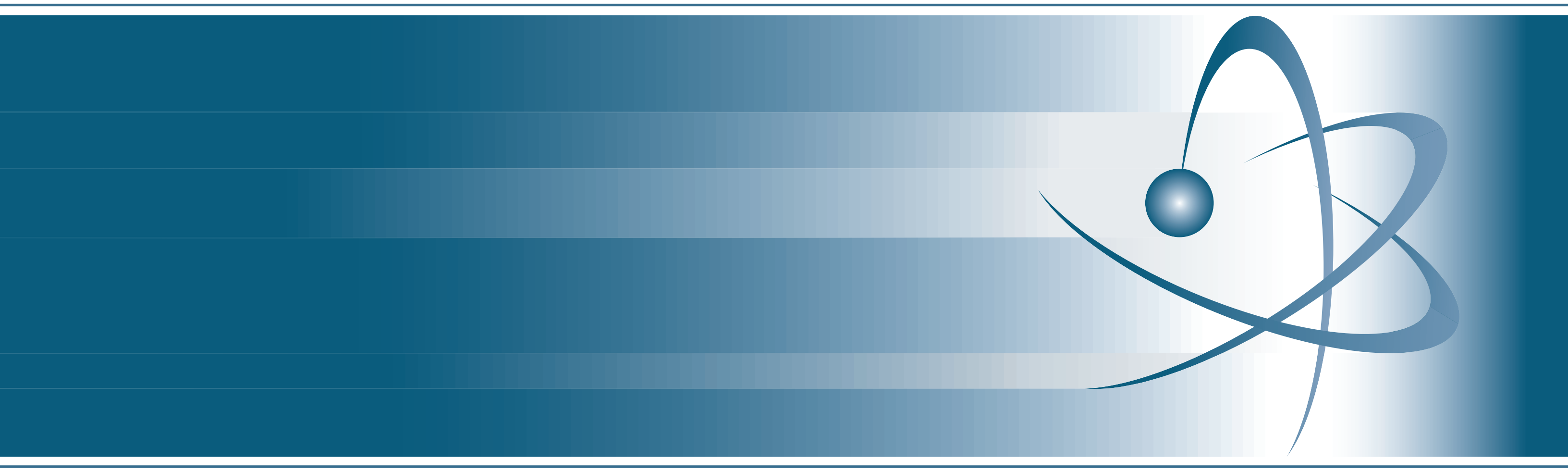
Responses to Questions Raised from Peer Review of Canada's Fourth National Report for the Convention on Nuclear Safety



Fourth Review Meeting
April 2008

Canada

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Responses to Questions Raised from Peer Review of Canada's Fourth National Report for the Convention on Nuclear Safety

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April 2008

This document supplements the Canadian National Report for the Fourth Review Meeting of the Convention on Nuclear Safety. By offering additional and detailed information in response to 123 specific questions received from 23 Contracting Parties, the document demonstrates how Canada has implemented its obligations under the Convention on Nuclear Safety. This document is produced by the Canadian Nuclear Safety Commission on behalf of Canada. Contributions to the document were made by representatives from Ontario Power Generation, Bruce Power, New Brunswick Power Nuclear, Hydro-Québec, Natural Resources Canada, Foreign Affairs and International Trade Canada, Atomic Energy of Canada Limited and Geological Survey of Canada.

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GENERAL COMMENTS					
1	Hungary	General	D.4, p.9	Section D.4 introduces that two organizations submitted application for licences to prepare sites for the future construction of NPPs. Q. How long does it take to issue the licences to prepare sites?	The main factor in the timing to issue a licence to prepare a site is the duration of the environmental assessment (EA), which must be conducted in accordance with the *Canadian Environmental Assessment Act* (CEAA) (see Attachment 1 for relevant excerpts from the CEAA). The present planning assumption is that the EA may take up to three years. A regulatory review of an application for a licence to prepare a site may be performed concurrent with the EA. Thus, such a licence may be issued shortly after the completion of the EA.
2	Hungary	General	D.4, p.9	Section D.4 introduces that two organizations submitted application for licences to prepare sites for the future construction of NPPs. Q. What kinds of reactors are proposed?	Bruce Power proposed five designs: ACR-1000, AP1000, EPR, ESBWR and Enhanced CANDU-6 (EC-6). Ontario Power Generation proposed nine designs: EC6, EPR, AP1000, APWR, OPR1000, APR1400, ABWR, ESBWR and ACR-1000. Subsequently, on March 7, 2008, the Ontario Ministry of Energy announced that they have started the two-phase competitive request for proposal (RFP) process to select a nuclear reactor vendor. The following four international vendors have been invited to participate in the RFP process: AREVA NP, AECL, GE Hitachi Nuclear Energy, and Westinghouse Electric Company.
1B	Argentina	General		The report states that Bruce Power and Ontario Power Generation submitted applications for new NPPs to CNSC. Could Canada provide details about the type of reactor will be licensed, and the CNSC provision if the reactors are different from CANDU type?	Bruce Power proposed five designs: ACR-1000, AP1000, EPR, ESBWR and Enhanced CANDU-6 (EC-6). Ontario Power Generation proposed nine designs: EC6, EPR, AP1000, APWR, OPR1000, APR1400, ABWR, ESBWR and ACR-1000. Subsequently, on March 7, 2008, the Ontario Ministry of Energy announced that they have started the two-phase competitive request for proposal (RFP) process to select a nuclear reactor vendor. The following four international vendors have been invited to participate in the RFP process: AREVA NP, AECL, GE Hitachi Nuclear Energy, and Westinghouse Electric Company. Since it is not the role of the regulator to influence the design to be built (by expressing either a positive or negative opinion), the Canadian Nuclear Safety Commission (CNSC) is adopting a technology-neutral approach. This is exemplified by the draft CNSC regulatory documents RD-337, *Design of New Nuclear Power Plants*, and RD-346, *Site Evaluation of New Nuclear Power Plants*, which draw from IAEA standards NS-R-1 and NS-R-3, respectively. The remainder of the regulatory framework to be prepared will also be technology neutral, as will the assessment plans and review guides that the CNSC is preparing. To this end, the CNSC is putting resources into becoming familiar with specific aspects of light water reactor technology.
3	Romania	General	D4	During the reporting period, two licensees	A total of nine designs are proposed by the two proponents: Enhanced CANDU-6, EPR, AP1000, APWR, OPR1000, APR1400, ABWR, ESBWR and ACR-1000. Subsequently,

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				(Bruce Power and OPG) submitted applications to the CNSC to build new NPPs. Please provide some details about the plant design decided by these companies to build.	on March 7, 2008, the Ontario Ministry of Energy announced that they have started the two-phase competitive request for proposal (RFP) process to select a nuclear reactor vendor. The following four international vendors have been invited to participate in the RFP process: AREVA NP, AECL, GE Hitachi Nuclear Energy, and Westinghouse Electric Company. Detailed information about the specific designs being proposed is not available at this time, because it is not required under Canadian regulations until an application for a licence to construct is made.
4	India	General	Page 15, section III C	It is reported that the calandria tubes at Bruce A Units 1 and 2 are being replaced. In addition to the sag of the calandria tubes, are there any other considerations for deciding on their replacement. What is the designed life of the new calandria tubes?	The primary reason for calandria tube (CT) replacement is sag and the expected difficulty in installing straight pressure tubes into sagged CTs. As well, some sagged CTs will eventually approach and contact shutdown system 2 liquid injection nozzles, which would result in through-wall fretting of the CTs over time. Additional consideration was given to improvements in material qualities. The design life of the replacement CTs is estimated at 30 years.
5	Pakistan	General	D.3, Page 8	Please refer to fourth paragraph of section D.3 on page 8, it is stated that material conditions of steam generator of unit 2&3 were found to be much worse as compared to unit 1 & 4. Please explain what were the reasons for the difference in degradation when all the steam generators were functioning under same operating conditions?	The condition of Pickering A's steam generators in units 2 and 3 was a significant factor in the decision not to restart the units, but not the sole factor. For Unit 2 steam generators, inside diameter intergranular attack (ID IGA) is the most probable life-limiting condition. A root cause investigation concluded with high confidence that the initiating event was produced during an off-line decontamination operation. Unit 1 also has significant ID IGA degradation, also believed to be from the same off-line decontamination operation. Unit 4 has only a small quantity of suspected ID IGA. The ID IGA degradation in Unit 2 is much more severe compared to that of units 1 and 4. Only 5 of the 12 steam generators in Unit 3 were inspected with the basic probe. Results indicated that there are a large number of heavily dented (deformed) tubes in all five steam generators and one steam generator with a large number of ID indications (possibly IGA, but not confirmed). Also, the Unit 3 steam generator 5 hot leg tubesheet has severe damage from a loose part. Unit 1 has some significant denting, and a secondary side chemical clean was performed to mitigate future denting. Unit 4 has almost no denting. Whether denting occurred during initial manufacture or was different between units due to operating chemistry differences is not known. Prorated based on inspection of five steam generators, the amount of denting in Unit 3 is much greater than in Unit 1.

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6	Russian Federation	General		CANDU reactor like all other channel-type reactors is a multipurpose reactor as is mentioned in the Report when speaking about isotope production. How do you deal with the issue of possible multipurpose CANDU application, when the reactor is delivered abroad?	CANDU is not designed as a multipurpose reactor. The general statements in Section A of Chapter II, on page 5, of Canada's Fourth National Report relate to numerous Canadian nuclear technologies, including that of the CANDU reactors. These statements may have led to the interpretation that CANDU reactors are used for the production of medical isotopes; this is incorrect. Isotopes for medical use are produced in non-power reactors that are licensed for such activities. Notwithstanding these general clarifications, Pickering B; Bruce B; and Gentilly-2 are authorized to produce Cobalt-60 as a by-product from adjuster rods containing cobalt. This is similar to authorizing the removal of tritium from tritiated heavy water used in CANDU reactors. CANDU reactors are subject to comprehensive and rigorous safeguards agreements and additional protocols (not covered by this Convention), both in Canada and abroad.
7	United States	General		Does Canada envision any diversification of power reactor designs other than the CANDU should new power plants be constructed? If so, how is the CNSC preparing for licensing and regulatory oversight activities involving a new design?	At present, nine designs are being proposed by the two Ontario proponents: Enhanced CANDU-6, EPR, AP1000, APWR, OPR1000, APR1400, ABWR, ESBWR and ACR-1000. Subsequently, on March 7, 2008, the Ontario Ministry of Energy announced that they have started the 2-phase competitive Request For Proposal (RFP) process to select a nuclear reactor vendor. The following four international vendors have been invited to participate in the RFP process: AREVA NP, AECL, GE Hitachi Nuclear Energy, and Westinghouse Electric Company. Since it is not the role of the regulator to influence the design to be built (by expressing either a positive or negative opinion), the Canadian Nuclear Safety Commission (CNSC) is adopting a technology-neutral approach. This is exemplified by the draft CNSC regulatory documents RD-337, *Design of New Nuclear Power Plants*, and RD-346, *Site Evaluation of New Nuclear Power Plants*, which draw from IAEA safety standards NS-R-1 and NS-R-3, respectively. The remainder of the regulatory framework to be prepared will also be technology neutral, as will be the assessment plans and review guides that the CNSC is preparing for the various approvals or licences which must be given. To this end, the CNSC is putting resources into becoming familiar with specific aspects of light water reactor (LWR) technology, and regulatory requirements and practices in various countries in which LWR were licensed. Once the design (or designs) to be built is selected, the CNSC will initiate the development of lower-level, technology-specific guidance documents.
8	United States	General		The report describes a 12-month pilot application of risk-informed decision making which ended in	There were lessons learned from the pilot program. The lessons emphasize, among other things, the importance of the following: - delivering adequate training on the use of the process; - making proper team selection; - correctly identifying and agreeing on the issue at hand;

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				May 2007. Were any lessons learned from the pilot program?	- ensuring that data and information used are accurate and current;, and - conducting adequate consultation with stakeholders. For additional information on the development and use of the risk-informed decision making process, please see Attachment 2.
ARTICLE 6: EXISTING NUCLEAR POWER PLANTS – nil					
ARTICLE 7: LEGISLATIVE AND REGULATORY FRAMEWORK					
9	Australia	7.1		Australia is keen to learn of the indicators that the CNSC might use to measure the effectiveness and performance of its nuclear safety regulatory framework. For example, we have an interest in indicators used to measure: - the effectiveness of outcomes and processes; - efficiency of processes in terms of timeliness, cost and resource utilisation; - effectiveness of enforcement and compliance activities; and - stakeholder satisfaction.	The Canadian Nuclear Safety Commission (CNSC) has developed indicators for measuring its effectiveness and performance that are tied to its core activities. These indicators are routinely reported in the CNSC's annual report to the Government of Canada. The CNSC is further working on its overall performance management program to improve on its existing indicators and link them to key regulatory processes. Examples of performance indicators include the degree of progress (Excellent, Good, Appropriate) of planned priorities; level of meeting the strategic outcome of program activities against established measures; and extent of deviation between planned and actual spending. Specifically, the CNSC produces the following two reports annually: 1. The *Departmental Performance Report* (DPR), which provides a focus on results-based accountability by reporting on accomplishments achieved against the performance expectations and results commitments as set out in the *Report on Plans and Priorities*; and 2. The *Report on Plans and Priorities* (RPP), which provides increased levels of detail on a business line basis and contains information on objectives, initiatives and planned results, including links to related resource requirements over a three-year period. The RPP also provides details on human resources requirements, major capital projects, grants and contributions, and net program costs. These reports are tabled in Parliament by the President of the Treasury Board on behalf of the ministers who preside over these organizations. Additional information as well as the above-mentioned reports are available on the CNSC Web site at www.nuclearsafety.gc.ca
10	Australia	7.1		With regard to the issue of transparency in nuclear safety regulatory decision making,	The Canadian Nuclear Safety Commission (CNSC) is an independent agency that operates in a transparent manner. When establishing regulatory policies and making licensing decisions, the "Commission Tribunal" (simply referred to as the Commission) considers the views, concerns, and opinions of interested parties and intervenors. The *Nuclear Safety and*

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				Australia would be grateful for any information that Canada could provide on the processes it has in place to achieve transparency of the decision making process, for both licensees and members of the public, particularly where there is no legislated process in place.	*Control Act* (NSCA) requires that the Commission hold public hearings for most licensing matters. In addition to notifying the applicant or licensee, the Commission publishes all notices of public hearings 60 days in advance. Members of the public or intervenors may participate by attending in person or have their written submissions considered in a public forum. In addition to using its public hearing room in Ottawa, the Commission periodically conducts hearings at specific locations to afford greater opportunity for engaging members of the local public. The Commission also uses, where appropriate, teleconferencing, videoconferences and video webcasting. For additional and detailed information, please see Attachment 3
11	China	7.2.1	CH‡W 7.2	How did CNSC choose a technical support organization?	The Canadian Nuclear Safety Commission (CNSC) does not use a technical support organization as other regulators. When external technical support is required, the CNSC contracts out work to the private sector, universities or other agencies and organizations in Canada and elsewhere. Contracts are placed in accordance with the Treasury Board of Canada's Contracting Policy. Contracts are usually issued following a competitive process; however, the policy allows for contracts to be directed to a specific contractor under certain circumstances. For a competitive contract, a request for proposal (RFP) is placed on the Government of Canada's open bidding service,) which is an electronic bulletin board used to advertise government needs. It is currently available on the Internet at the MERX Web site. Bids received in response to the RFP are evaluated and the contract is awarded to the winning bidder. The CNSC evaluates bids using the following criteria: a) Technical • understanding of scope of objective; • recognition of direct as well as peripheral problems and solution offered; • proposed approach and methodology; and • adequacy of work plan and schedule. b) Personnel • project manager (relevant experience, qualifications, position within the organization); • key personnel (relevant experience, qualifications); and • team organization planned. c) Company experience • competence proven by similar and/or related work;

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				(page. 59) In this sentence, what is the method and indicator (or criteria) to assess and judge the employee performance?	standards of performance used in the observation and coaching (for example indicator, criteria) are those written procedures that directly specify the particular work task (for example, maintenance procedure for overhauling a pump) and those general procedures widely applicable (for example, safe work practices, use of personal protective equipment, housekeeping expectations).
53	Korea, Republic of	12	Section 12 b	(Article 12, Section 12 b) In the page 60, "Methods to Prevent, Detect and Correct Human Errors" states that HFE is applied in new designs. What are the major items in new designs which the HFE is applied to?	Human factors engineering (HFE) considers operational, maintenance and decommissioning tasks. Considerations are included in all modifications to existing plants, for plant life extensions and for new builds. HFE effort increases with higher levels of interface complexity or criticality, and more HFE effort is typically required for operator tasks. Examples of common applications of HFE principles in new designs are in the selection of human system interface components, equipment layouts, control room habitability, control room display design, panel design and annunciation design. For example, each of these aspects would typically factor in changes when switching from analog to digital technology. These changes could be at the component or system level.
54	Switzerland	12	Pages 59 & 61, paragraphs 12a and 12c	What are CNSC's policies concerning Human Factors competencies of its personnel? Who is entitled to perform oversight activities regarding Human Performance (e.g. assessment of Human Factors Programmes)?	At the Canadian Nuclear Safety Commission (CNSC), senior human factors specialists are expected to possess at least a master's degree in human factors engineering, industrial engineering, engineering psychology, ergonomics or other related degree. Most of the CNSC's seven senior human factors specialists possess a Ph.D. in human factors. In addition, it is desirable for such specialists to have in-depth relevant experience in a process industry. They are also expected to have knowledge in a variety of disciplines such as human factors principles, theories, methods, standards, and guidelines; human cognitive and physical capabilities and limitations; and human-machine and human-computer interface design and assessment. Please see Attachment 6 for detailed information on the CNSC Human Factors staff competency.
55	United Kingdom	12	Page 61	This section of the report refers to concerns over compliance with the limits on hours of work of NPP staff. Under Article 19, Section 19 (iv), page 107, the report states that "An operating licence condition specifies the minimum staff complement that	Minimum shift complement requirements and adherence to limits of hours of work are both required of licensees. A minimum shift complement violation must be corrected immediately so the station has an adequate number of trained personnel at all times. Licensee staff must also adhere to the limits of hours of work defined in their internal procedures. The Canadian Nuclear Safety Commission (CNSC) has provided guidance on limits to hours of work so licensees can develop programs to manage the long-term cumulative effects of fatigue on work performance. Facility management staff may face two sets of requirements that can not be met concurrently. For example, staffing levels in a certain position may not be met at the start of a given shift. It is required that minimum shift complement be maintained at all times,

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				must be present at the station at any time." Does this represent a situation in which the requirement to comply with the requirement on minimum staff numbers is leading to possible breaches in the limits on hours of work? How does CNSC intend to ensure that neither requirement is compromised? Does CNSC plan any special measures to ensure that "the increasing reliance on contracted staff at the NPPs" does not jeopardise nuclear safety?	and this would take priority over adherence to hours of work limits. Qualified staff already in the station would be required to stay until relief was arranged. The short term effects of fatigue on worker performance could be managed. Paragraph 12 (1) (a) of the *General Nuclear Safety and Control Regulations* requires every licensee to "ensure the presence of a sufficient number of qualified workers to carry on the licensed activity…" The CNSC monitors staffing levels and adherence to hours of work limits as indicators of licensee ability to meet this requirement. If it is noticed that staffing levels are low and that hours of work violations are increasing, then the CNSC would direct the licensee to address the issue and create a long term plan for improving overall staffing levels. Given the current work and expected growth of the nuclear industry, the CNSC anticipates the use of contract staff to continue increasing in the coming years. Licensees maintain overall responsibility and accountability for safe facility operation. The CNSC has and will continue to inspect licensee contractor management programs to ensure that licensees adequately oversee work performed by contractors. Human factors specialists are particularly interested in ensuring that contractors working in a nuclear facility are properly qualified, supervised and trained in station processes and error prevention strategies. The CNSC is working with licensees to establish provisions for the limits of hours of work for contract personnel. Discussions are also underway to document the types of work that contract staff are permitted to do, and the potential effect this work will have on nuclear safety.
ARTICLE 13: QUALITY ASSURANCE					
56	Slovenia	13	p. 64	The QA program is binding on all personnel whose work on the nuclear project can affect nuclear safety. This includes the work performed by organizations that are not part of the licensee's organization. The requirements of the CSA-N286 series apply also to	Canadian licensees ensure that all external organizations that are performing work that can affect nuclear safety have a Quality Assurance (QA) program and a corresponding certificate. Assuming that these organizations do have a QA program, a copy of their certificate and program manuals are procured. Licensee procurement and vendor quality assurance Specialists compare these manuals against Canadian Standards Association (CSA) Standard Series CSA-N286 to ensure that all aspects of this series are captured in their programs. Also, as members of the CANDU Procurement Audit Committee and the Nuclear Procurement Issues Committee, which provide a cost- and quality-effective program for evaluating suppliers that furnish nuclear safety-related items and services, Canadian licensees can audit external organizations or have a third party audit them. Licensees have access to previous audit reports to measure an organization's adherence to its QA standard and to compare it against CSA Standard Series CSA-N286. If aspects of

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				the work performed by organizations that are not part of the licensee's organization. How do you treat the organizations with the QA programs that are not in accordance with the CSA-N286 (ISO 9001:2000, 10 CFR 50 App. B, IAEA QA Safety Series, ASME code, etc.).	this series are not reflected in programs of the external organization, a corrective action is raised requesting programs to be altered accordingly.
57	Switzerland	13	Page 66, paragraph 3	How is IAEA GS-R-3 considered in national requirements today?	IAEA GS-R-3 is considered both in the requirements being applied within the regulatory body and in the establishment of requirements applicable to nuclear facilities and activities. For example, the Canadian Nuclear Safety Commission (CNSC)'s corporate-wide management system takes into account the requirements specified in GS-R-3 and their applicability to regulatory activities. Furthermore, the CNSC is completing regulatory guidance documents that will guide nuclear facility operators and nuclear substance users to align their existing management systems with IAEA GS-R-3.
ARTICLE 14: ASSESSMENT AND VERIFICATION OF SAFETY					
58	Euratom	14.1	Section 14 (i) e, p. 72	Can you clarify the differences between an IRS [ISR] conducted for a reactor re-start and for a reactor life extension?	Licensees are expected to follow the IAEA Safety Guide NS-G-2.10 on periodic safety review (PSR). The key difference between a review for a re-start versus one for life extension is the context for decision-making. For restart, decisions need to be made in view of the remaining operating life (for example, five years), whereas for life extension, the decision need to consider proposed long-term operation (for example, 25 to 30 years). In both cases, the licensee needs to demonstrate how the plant, including systems, structures and components, will safely be operated and maintained during the proposed operating life. Note that the decision is not tied to the "typical" 10-year timeframe used in PSR applications. In addition, an Environmental Assessment (EA) would most likely be performed for life extension. In such a case, the findings of the EA along with those of the PSR (or the equivalent Integrated Safety Review) will form the bases for developing the integrated implementation plan of corrective actions and safety improvements.
59	Finland	14.1		International cooperation for regulatory related	Canada shares the view that nuclear safety research is important in supporting safe plant design and operation. In Canada, it is the responsibility of a plant designer and/or licensee

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				nuclear safety research is an important issue to be considered. What is your view or opinion concerning the needs in your country for large nuclear safety related experimental test programmes to study physical phenomena and to validate analysis models used in safety analysis (e.g. three dimensional reactor physics and thermal hydraulic models etc)? Are such experimental research and analysis work needed for safety upgrading or assessment of safety in case of periodic safety review or plant life extension in your country or for new reactors?	to provide adequate safety justification in order to obtain licensing approval. Fulfilling this responsibility includes provision of adequate experimental data to support analytical models and safety analyses. As practice shows, ongoing experimental research is needed for operating plants as well as for plant life extension and new reactors. The need for experimental research was further emphasized by a recently completed project by the Canadian Nuclear Safety Commission (CNSC) that led to the development of a risk-informed position on outstanding safety issues with focus on the design and safety analysis. This risk-informed position is of particular importance to focus research efforts on safety-significant areas, and to facilitate the development of plant-specific safety improvement programs (to support plant re-licensing and life extension projects) or the reviews of new reactor designs. The industry R&D programs are coordinated and managed by the CANDU Owners Group, with current funding of about $38 million annually. The CNSC also maintains a Research and Support Program (RSP) with the mandate of generating knowledge and information to support CNSC staff in its regulatory mission. With an annual budget between 2 and 3 million dollars, the RSP enables CNSC staff to engage the services of external experts and experimental facilities when needed to support regulatory decisions and assess emerging safety issues.
60	France	14.1	P. 72, § 14(i)d	Could Canada clarify if PSA results were used during Integrated Safety Reviews (ISRs) (for identifying and/or setting priorities to plant safety improvements?) In a possible PSR process, is the use of PSA planned?	Use of probabilistic safety assessment (PSA) results is considered as an integral part of the integrated safety review (ISR). For Point Lepreau, the Canadian Nuclear Safety Commission required a PSA to be completed in conjunction with the refurbishment outage, in order to use the results of the PSA to establish the refurbishment scope. Preliminary PSA results led to some additions to plant modifications. For Bruce A and Pickering A refurbishment projects, PSA results were used to identify plant safety improvements.
61	Germany	14.1	Section 14 (i) b, Pages 69-70,	The definition of generic safety issues/Generic Action Items (GAI) for	Canada appreciates this assessment. Indeed, in our view, the Generic Action Items offer a systematic process to address those safety issues that apply to several operating facilities and often require experimental investigation.

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			Appendix	safety verification to be concentrated upon is applied in Canada. These GAI are deduced, for example, from generic Risk-Informed Decision Making (RIDM) processes, severe accident analyses, etc. The Canadian utilities are asked to show compliance with these GAI, rank their importance and develop a strategy of coping with the safety items. Appendix F gives a survey and outlines deliverables to be presented by a utility for the related closure of the GAI. The outlined concept represents a good and promising practice. It is recommended for countries with reactors of similar types.	For additional information on the development and use of the risk-informed decision making, please see Attachment 2.
62	Korea, Republic of	14.1	Appendix F	(Article 14-1, Appendix F) Please provide following information - the quantitative or qualitative definition of severe core damage and large off-site radioactive release for CANDU reactor.	Severe core damage is defined for CANDU designs as the failure of two or more fuel channels in the core. The Canadian Nuclear Safety Commission draft regulatory document RD-337, *Design of New Nuclear Power Plants* defines a large off-site release as "a release of radioactive material which could require long-term resettlement of the public in order to prevent unacceptable health effects as a result of severe core damage and failure of containment. The corresponding frequency is defined as the "sum of all event frequencies that can lead to release of more than $10E^{14}$ Bq of Cs_{137}", with a target of $10E^{-7}$ and a limit of $10E^{-6}$ per plant per year. Regarding estimates of hydrogen release into containment, Chalk River Laboratories has

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				- detailed information on hydrogen behavior and amount to be released to the containment during severe core damage. (This question is related with the reports on GAI 88G02 "hydrogen behavior in CANDU nuclear generating plants")	done preliminary scoping analyses for some severe accidents for CANDU-6 designs. The MAAP4-CANDU code was used to conduct these analyses.
63	Pakistan	14.1	Article 14(i) b, Page 69	It is observed that a number of GAI's were found to be "open" till January 2004 concerning OPG, Bruce, Hydro Quebec and NBPN, how many have been closed during 2005-6. It is reported that CNSC commenced work on a project for ranking these issues on the basis of importance; and developing a strategy to resolve these issues in the context of new NPPs as well as those in operation or being refurbished. What criterion and strategy is being followed for ranking these GAI's	Out of the five industry-wide Generic Action Items (GAIs) reported as closed since January 2004, there were three that were closed in 2005–2006. For other GAIs, closures have been requested and the supporting information is under review by Canadian Nuclear Safety Commission (CNSC) staff. In 2007, the CNSC completed a project to prioritize the known safety issues, including GAIs and design and safety analysis issues identified in the IAEA TECDOC 1554, *Generic Safety Issues for Nuclear Power Plants with Pressurized Heavy Water Reactors and Measures for their Resolution*. To rank the issues, criteria defined in the risk-informed decision making (RIDM) process were applied, with consideration of the likelihood and consequences of scenarios where such issues may be of importance. As a result, all initially identified issues were placed into three categories: Category 1 – not a safety important issue for Canadian reactors; Category 2 – an issue, but appropriate measures are already in place; Category 3 – an issue that still needs resolution. Work is currently progressing with the Category 3 issues to implement adequate resolution For additional information on the development and use of the RIDM process, please see Attachment 2.
64	Pakistan	14.1	Article 14.1, Page 68	Reference Chapter-II "Context" section D.3 on page 8 and Section 14.1, page 68, it is stated that life of NPP's is extended	In accordance with the Canadian Nuclear Safety Commission Regulatory Document RD-360 (that succeeded G-360), *Life Extension of Nuclear Power Plants*, utilities that are planning life extensions are required to carry out an integrated safety review (ISR) based on the IAEA periodic safety review (PSR) guide. A major part of the assessment is to determine the condition of safety-related structures, systems, and components. This

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				by replacing the fuel channel, how is the integrity and safe operation of other safety related structures and systems assessed with regards to ageing ?	condition assessment, which includes inspections and analysis, will determine to what extent some components require replacement. For components that will not be replaced, the assessment is used to update or develop life cycle management plans that will monitor the component condition, to ensure that it continues to meet its design function. For example, the integrity and safe operation of shutdown systems 1 and 2 is maintained through appropriate ageing management programs to satisfy regulatory testing requirements. In-core safety detectors have been replaced at various nuclear power plants in Canada as part of ageing management.
65	Romania	14.1		What is the licensing status of the ACR design in Canada?	Recently, the Canadian Nuclear Safety Commission (CNSC) decided to perform design reviews of new nuclear power plants under consideration for development in Canada, starting with the ACR-1000. Such design reviews will aid the CNSC in preparing for requests from proponents to review licence applications, in order to ensure a timely and transparent licensing process.
66	Switzerland	14.1	Page 72, second to last para.	This paragraph and Annex 14 (i) d suggest that no seismic PSAs have been conducted yet for the Canadian NPPs, however, a "PSA-based seismic margin assessment" is under development for the Point Lepreau NPP. What are the main features of this "PSA-based seismic margin assessment", as opposed to a seismic PSA and what were the key arguments for selecting the "PSA-based seismic margin assessment" rather than a seismic PSA?	The PSA-based seismic margin assessment (SMA) follows the same procedure and steps as those of the seismic PSA, except for the treatment of seismic hazard information. Since the PSA-based SMA does not consider the seismic hazard explicitly, it does not produce severe core damage. Instead, the PSA-based SMA produces results such as the seismic capacity and random failure probability, given that seismic events occur. The major driving force to adopt the PSA-based SMA was the large uncertainty in the seismic hazard, on which several orders of differences exist among experts. Past seismic PSA experiences indicated that the dominant factor affecting the seismic-induced severe core damage frequency was the uncertainty in the seismic hazard, not the seismic capacity of the nuclear power plants. This finding made the decision-making process quite difficult; and, consequently, it was proposed to use the PSA-based SMA. When the seismic hazard information becomes available with some consensus among experts, the PSA-based SMA can easily be converted to the seismic PSA. Point Lepreau is performing a PSA-based SMA for the purpose of refurbishment that uses existing PSA models and assigns a high confidence of low probability of failure to equipment, instead of a seismic fragility curve. The PSA-based SMA is defined in the US Nuclear Regulatory Commission document NUREG-1407.
67	Turkey	14.1	7.2.(i), P.23	Could Canada give more information about the design review process for	Early in 2008, and in preparing for requests from proponents to review licence applications, the Canadian Nuclear Safety Commission (CNSC) decided to perform design reviews of new nuclear power plants (NPPs) under consideration for development in Canada, starting

#	Country	CNS Article	Report Reference	Question	Answer
				new NPPs?	with the ACR-1000. Furthermore, the *Nuclear Safety and Control Act* requires a proponent to submit an application for a licence to construct that includes plant design details along with a preliminary safety analysis report. However, an Environmental Assessment (EA) is conducted in support of an application for a licence to prepare a site, prior to or concurrent with the application for licence to construct. During the EA process, CNSC staff must be able to review general descriptions of the proposed design(s) in order to determine the environmental impacts on the site and the surrounding area. Upon receipt of an application, a corresponding assessment plan is developed. When this plan is approved, a CNSC point-of-contact is named to coordinate an integrated review of technical and legal aspects, with the goal of confirming that regulatory requirements are satisfied. Supplementary information on design review process for new build is available on the CNSC Web site at www.nuclearsafety.gc.ca.
68	Turkey	14.1	14 (i).f, P. 75	What is the expected date for the clarification of the CNSC's decision related to the adapting of PSR methodology into the Canadian licensing regime?	In the past year, the Canadian Nuclear Safety Commission (CNSC) began a project to review licensing requirements and practices for nuclear power plants. The scope includes the introduction of requirements for formal periodic safety reviews (PSRs). Work completed to date includes participation in IAEA technical meetings, reviews by expert consultants, consultation meetings with industry, and development of proposed licence conditions. In the near future, presentations are planned for the Commission's information and decision on PSR, including outlining the transition to a PSR methodology, which is expected to take several years to implement.
69	United Kingdom	14.1	Page 72	In relation to Integrated Safety Reviews (ISRs), the report says "ISRs are being performed in support of reactor re-start and life extension projects." Could CNSC please provide details for each operating reactor of the dates on which either of these triggering events has led to an ISR being performed? Given this requirement to perform ISRs, what might be the	Ontario Power Generation (OPG) initiated an integrated safety review (ISR) for Pickering B reactors in June 2006. Bruce Power initiated an ISR for Bruce units 1 and 2 in 2005. For Gentilly-2, a "Revue de sûreté" was initiated in 2001, which Hydro-Québec considers as equivalent to an ISR. New Brunswick Power Nuclear initiated an ISR for Point Lepreau in 2000. The ISR is a one-time application of the IAEA periodic safety review methodology, in view of long-term operation of the facility. Currently, there is no requirement to perform successive ISRs. However, the current licensing process in Canada involves frequent re-assessment to support licence renewals, including consideration of modern standards. These licence renewal re-assessments could involve some or all the factors comprising an ISR.

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				longest period between successive IRSs on each reactor? How does the ISR process take account of the need for a comparison with modern standards, and an analysis to see whether updating the plant to meet modern standards is reasonably practicable?	
70	India	14.2	Page 128, section A	In response to GAI 95G01: "Molten Fuel/Moderator Interaction", several actions on the part of the licensee and the regulator have been reported for arriving at the closure criteria. Please elaborate on the closure criteria proposed by the licensee and any additional criteria specified by CNSC.	Closure criteria for GAI 95G01, "Molten Fuel/Moderator Interaction" (MFMI), are identified in the position statement developed by Canadian Nuclear Safety Commission staff. This includes identification of the dominant mode of the MFMI following fuel Channel failure at high pressure. Licensees are also expected to utilize test results to evaluate the safety margin or potential damage resulting from the MFMI.
71	India	14.2	Page 74, section 14 (i) e	In Pickering A Unit 1 & 4 extensive upgrades were done, including major enhancements to shutdown system capability and improvements to the emergency core cooling systems. Please indicate the engineering changes carried out to enhance shut down system	For Pickering A units 1 and 4, a shutdown system enhancement (SDSE) was added to the existing Pickering A shutdown system (SDS) A. To the extent practicable, the SDS A and SDSE were made independent of each other. The enhancement provided a new set of triplicated trip sensors and trip logic augmented with new moderator dump logic. The enhancement also included the addition of two shutoff rods, bringing the total number to 23. In addition, modifications were completed in the emergency coolant injection (ECI) system to reduce the predicted severe core damage frequency, and to perform system upgrade to meet environment qualification, seismic requirements and other system improvements. For additional and detailed information, please see Attachment 7.

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				capabilities & improvements in the ECCS.	
72	India	14.2	Page 72, section 14 (i) c	It is reported that the fuel channel diametral creep was one of the causes for de-rating of units at Point Lepreau and also at other places. What are the ranges and rates of diametral creep observed till date? Up to what diametral creep, the units are permitted to operate without de-rating?	Pressure tube (PT) diametral creep is just one aspect of heat transport system ageing that impacts cooling of fuel. Other parameters that must be considered include steam generator fouling, heat transport piping roughening, flow losses, pressure drop changes in system, and increases in reactor inlet header temperature over time. The degree of diametral creep permitted without derating is dependent on a number of design conditions, design assumptions, and actual plant conditions. Hence, the time at which derating is required is unit specific and depends on the operating history and current operating conditions. The range of diametral creep observed to date is also dependent on the pressure tube material installed and the conditions to which the material is exposed, with the most influential parameters being fast neutron fluence and irradiation temperature. For CANDU-6 reactors, it has been determined that critical channel power penalties have to be applied for PT creep larger than about 2.1%.
73	India	14.2	Page 8, section Chapter II D.3	In Pickering A, the steam generators in Units 2 and 3 were reported to be in much worse condition than those in Units 1 and 4. Whether any reasons have been identified for poor conditions of SGs of Pickering units 2&3 as compared to those in units 1 & 4.	The condition of Pickering A units 2 and 3 steam generators was a significant factor in the decision not to restart the units, but not the sole factor. For unit 2 steam generators, inside diameter intergranular attack (ID IGA) is the most probable life limiting condition. A root cause investigation has concluded with high confidence that the initiating event was produced during an off-line decontamination operation. Unit 1 also has significant ID IGA degradation also believed to be from the same off-line decontamination operation. Unit 4 has only a small quantity of suspected ID IGA. The ID IGA degradation in Unit 2 is much more severe compared to that of units 1 and 4. For Unit 3 steam generators, only 5 of 12 steam generators were inspected and only with the basic probe. Inspection results indicated several heavily dented (deformed) tubes in all five steam generators and that one steam generator contains a large number of ID indications (possibly IGA, but not confirmed). Also, the steam generator 5 hot leg tubesheet in Unit 3 has severe damage due to a loose part. Unit 1 has some significant denting and a secondary side chemical clean was performed to mitigate future denting. Unit 4 has almost no denting. Whether denting occurred during initial manufacture or varied between units due to operating chemistry differences is not known. Prorated based on inspection of five steam generators, the amount of denting in Unit 3 is much greater than in Unit 1.
74	India	14.2	Page 129,	It is reported that Bruce	Bruce Power revised operating procedures to enable alternate heat transport system (HTS)

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			section B	Power has made a number of improvements at Bruce A and B to reduce the risk associated with the postulated event of Pressure Tube Failure with Consequential Loss of Moderator and has plans to carry out further design modifications during plant refurbishment and fuel channel replacement. What were the improvements carried out and further modifications planned by Bruce Power.	make-up (gravity emergency core injection from the dousing tank) and alternate emergency coolant recirculation (D20 Recovery) for the subject event. This has significantly reduced the associated risk. Bruce Power has also revised operating procedures to enable rapid cooldown and depressurization of the HTS for the subject event. This reduces the probability of consequential calandria tube failure, thereby providing a small reduction in risk associated with the subject event.
75	Pakistan	14.2	Article 14(ii) d, Page 78	The CNSC uses five rating categories to assess licensee programs and their implementation in nine designated safety areas each encompassing one or more programs used by licensees and the CNSC to assess the safety of NPPs in Canada.. A summary of the ratings of all Canadian NPPs for the years 2003 through 2006 given in Table G.3 indicates that all Canadian Licensees fall in rating category “A” in the safety area Emergency Preparedness. Please provide	(a) Emergency preparedness programs are updated and fine-tuned over the life of the facility as new requirements are identified or to handle changing conditions or identified deficiencies. Notwithstanding the fact that the programs have matured and are well-maintained, Canadian Nuclear Safety Commission (CNSC) staff has observed that power reactor operators in Canada proactively seek ways to continuously improve their emergency preparedness programs. At this point in Canada, the nuclear power plants have all reached a level of maturity and have achieved a grade of “A” for their emergency preparedness programs. (b) Various safety areas and programs are rated in terms of CNSC staff expectations (see Attachment 4 for details).

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				information on (a) what factors are contributing to such performance (b) Based on the CNSC standardized nine safety areas, how is the 'overall' safety assessment /ranking carried out?	
76	Russian Federation	14.2	Ageing Management Plans.	Do you have a substantiation of the scope and intervals of reactor material/structure inspections with due account of the ageing rates and reliability of inspection data?	As a standard practice fitness-for-service methodologies, sanctioned by the Canadian regulator, require a prediction of the reactor material/structure/component degradation over the postulated operating interval in terms of: a. predefined inspection scope, including provisions for potential discovery based expansion scope that is specific to the degradation mechanism of concern; b. rate of degradation over the next operating interval; c. relevant material properties accounting for changes to these properties; and d. the reliability of the inspection techniques utilized to establish the system/component condition at the time of the inspection. In addition and where relevant, all of these inputs are conservatively pro-rated to the end of the operating interval, and these analysis-results are compared with applicable safety factor-derived acceptance criteria from the applicable code/standard. If these acceptance criteria are not satisfied, the postulated operating interval duration is reduced and the fitness-for-service exercise is iteratively repeated to derive an operating interval that meets acceptance criteria. The final unit operating interval is then determined by the most limiting system/component identified within the specific inspection/maintenance campaign.
77	Switzerland	14.2	Pages 78, 137, 143	Who is responsible for the ratings given in table G3? How are the ratings generated? Does the lowest rating in a given safety area during the year determine rating given in table G3 or is the rating in table G3 a (weighted) average of all ratings of the year? Is there a algorithm to	Ratings for program design and implementation are derived through a process whereby specialists and/or inspectors evaluate a number of review topics in each safety area and program and present their findings to the regulatory program directors for discussion and approval. The process is outlined in an internal work-instructions document. There is currently no prescribed algorithm for rolling up the evaluations of the review topics, such as using an average, weighted average or lowest score methodology. Reviewers instead seek to balance their findings in terms of licensee performance in meeting the stated performance objective in each of the safety areas and programs. Various safety areas and programs are rated in terms of Canadian Nuclear Safety Commission staff expectations (see Attachment 4 for details).

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				determine the rating of table G3?	
ARTICLE 15 : RADIATION PROTECTION AND ENVIRONMENTAL SURVEILLANCE					
78	France	15	p. 170, Annexe 15.d	Could Canada give some information about the DRL (limits): in particular why the DRL/unit is different between the different plants? Could Canada explain why DRLs for Gentilly-2 are based on 5 mSv ? Could Canada specify why iodine is not taken into account in the liquid effluent releases?	The Canadian *Radiation Protection Regulations* limit effective doses to 1 mSv per year for members of the public; and all licensees with significant releases of radionuclides are required to calculate the upper limit of releases called derived release limits (DRLs). Calculation of DRLs is based on the Canadian Standard Association methodology (CSA, N288.1, 1987) and other developments in radiation protection; for example, ICRP dose conversion factors. DRLs are unique to each facility, vary in values, and depend on several factors (assumptions, critical group characteristics, site specific data, etc). Calculation of DRLs can vary from simple to very complex. It is important to note that, while a DRL for one facility may be higher than another facility of similar design (for example, as result of a different environmental setting), this does not mean that a licensee with a higher DRL would release more of a contaminant to the environment than other licensees. The Canadian Nuclear Safety Commission regulatory regime is based on the “as low are reasonably achievable” (ALARA) principle, which systematically drives emissions to the lowest reasonable levels. The Gentilly-2 DRLs document was established in 1990, before the more recent public dose limit of 1 mSv per year was set in the year 2000. Revision of the Gentilly-2 DRLs document, based on the public dose limit of 1 mSv per year, is in progress and updated version is expected in the near future. In addition, monitoring data on airborne emissions and liquid releases to the environment remain below 1% of the DRLs. Iodine is included in the DRLs list for liquid releases. Radiological environmental monitoring includes liquid releases in various sampling media (for example, precipitation, surface water, groundwater and potable water). Laboratory analysis or measurements are performed for all radionuclides, including iodine. However, results obtained for iodine are below the detection limit, and, consequently, are not reported by the licensees.
79	France	15	p. 79, §15.b	The text presents 3 particular strategies to minimize the dose for workers. Could Canada specify the benefit in term of dose (mSv) which has been recorded?	The three strategies (radiological exposure permits, airborne tritium reduction, and source term reduction) are on-going activities at licensee facilities. It is not easy to obtain quantitative information for the purpose of determining their benefit in terms of dose reduction. However, as a component of each licensee’s ALARA program, they are providing evidence that these strategies are being implemented as a means to reduce dose to workers. In many cases, the licensees and the CNSC have observed improving dose trends in terms of specific job tasks, which can be tied back to dose reduction initiatives.

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				Could Canada specify what are the different zones of controlled area and are the associated characteristics?	It can be noted in the table provided in Annex 15c of Canada's Fourth National Report, "Occupational Dose Summary for 2001 to 2005" that doses at each site have remained steady or have increased from 2001 to 2005; and in the table "Collective Dose at Canadian Nuclear Power Plants" that the total collective dose has been increasing over the last few years. The increase in dose is attributed to Canada's aging nuclear power plants and an increase in related outage maintenance and refurbishment. Also, the increase in dose is attributed to feeder tubes inspection. Zoning used at the nuclear power plants is a classification system of areas according to their potential for contamination. • Zone 1 – a clean area which is not a radiological zone and may be considered the equivalent of a normal public access area. • Zone 2 – a radiological zone that is normally free of contamination but is subject to infrequent cross-contamination due to the movement of personnel and equipment from contaminated areas. • Zone 3 – a radiological zone which contains systems and equipment which may be sources of contamination. • Unzoned area – an outdoor location, building or structure within the protected area that has not been otherwise explicitly zoned.
80	Korea, Republic of	15	Annex 15c	(Article 15, Annex 15c) In relation to Annex 15c: Doses to personnel at Canadian Nuclear Power Plants, it is stated that the CNSC Radiation Protection Regulations reflect the ICRP 60 and workers at Canadian NPPs are restricted by dose limits of 50 mSv in any one year and 100 mSv in a five-year period. IAEA RS-G-1.1 recommend that where doses to an individual worker exceed 20 mSv/y, the management should take the necessary	The Canadian Nuclear Safety Commission (CNSC) regulations do not have a specific requirement for licensees to take corrective actions where doses to individual workers exceed 20 mSv/year. However, all of the nuclear power plants (NPPs) have set action levels above which they must report to the CNSC in accordance with CNSC *Radiation Protection Regulations*, Section 6. This section states that an "action level" means a specific dose of radiation or other parameter that, if reached, may indicate a loss of control of part of a licensee's radiation protection program and triggers a requirement for specific action to be taken. When a licensee becomes aware that an action level referred to in the licence for the purpose of this subsection has been reached, the licensee shall: (a) conduct an investigation to establish the cause for reaching the action level; (b) identify and take action to restore the effectiveness of the radiation protection program; and (c) notify the Commission within the period specified in the licence. The action levels set by the NPPs are site specific and have been reviewed and accepted by the CNSC and are referenced in their licence.

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				corrective steps. - There are two data of the Bruce A&B and one data of Gentilly-2 to exceed 20 mSv in the table of Annex 15c. Did you take the corrective actions? If you did, please explain those actions.	When the *Radiation Protection Regulations* came into force in 2000, licensees were legally required to comply with the action level requirement when applying for a new licence, a licence renewal, or a licence amendment of significant nature. In the case of Bruce Power, this process was initiated in January 2002, and the action level requirement was incorporated into their licence in 2004. According to the National Dose Registry, the maximum individual dose for Gentilly-2 in 2003 is currently listed as 19.20 mSv. The difference in dose from that of 23.27 mSv reported in the Fourth National Report can likely be explained by a change to a dose that occurred in 2003, but which was only recorded later. Since the action limit for Gentilly-2 was not exceeded, no regulatory action was required from Hydro-Québec.
81	Netherlands	15	Annex 15d	What is the reason for the deviating DRL for Gentilly-2 from the other DRLs (based on 5 mSv against based on 1 mSv)?	Gentilly-2, like other nuclear facilities in Canada, should have a derived release limit (DRL) calculated on the basis of the 1 mSv/year limit. At Gentilly-2 the licensee has not yet completed the administrative process necessary to modify its licensing documentation to reflect the new limit. Nevertheless, the licensee must meet, and is meeting, the 1 mSv/year limit set out in the Canadian Nuclear Safety Commission (CNSC) *Radiation Protection Regulations*. Furthermore, licensees are required to maintain doses as low as reasonably achievable (ALARA) which normally, and in the case of the Gentilly-2, are a small fraction of the 1 mSv/year limit. Action Levels are also established to ensure correction of any abnormal conditions well before the limit could be approached or exceeded. Therefore, while the DRLs have not been updated in a timely manner at Gentilly-2 (and this is being corrected) the CNSC is satisfied that the facility is operating and regulated in a manner that is protective of health and the environment and fully consistent with other similar facilities and the International Commission on Radiological Protection’s recommendations. For additional information on ALARA, dose limits, and action levels, please see Attachment 8.
82	Netherlands	15	Annex 15c	What is the reason for the relative high maximum individual doses compared to other countries? In Canada the maximum is in almost all cases well above 10 mSv while in some other countries there is a dose constraint of about 10	Canada faces the challenge of an ageing reactor fleet. To maintain a safe and effective operating state, maintenance work is necessary, and includes reactor restarts, extended maintenance, refurbishment outages, and feeder tube inspection and replacement. All of these tasks involve dose-intensive work that results in an increase in collective dose received. In Canada, the regulatory dose limits to workers are 50 mSv in a year and 100 mSv over five years. In addition, the licensee is required to keep the amount of exposure as low as reasonably achievable (ALARA), social and economic factors being taken into account. Furthermore, all nuclear power plants have action levels incorporated into their licence

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				mSv.	requirements, which are used as a means of indicating a potential loss of control of licensee radiation protection programs and trigger a requirement for specific action to be taken if exceeded. For additional information on ALARA, dose limits, and action levels, please see Attachment 8.
83	Netherlands	15	p.80	Do the NPPs in Canada use the principle of dose constraints for certain jobs?	Dose constraints are not a requirement under the Canadian Nuclear Safety Commission (CNSC) *Radiation Protection Regulations*. Dose targets for planned and emergent jobs are assessed for adherence the “as low as reasonably achievable” (ALARA) principle and entered into radiological exposure permits. Each job is assigned a target or projected dose that is discussed before performing work. These targets or projections are based on historical data, dose reduction initiatives, and current field survey information. The licensee works to perform each job with the final collective dose at levels equal to or below the projected target. Furthermore, all nuclear power plants have action levels incorporated into their licence requirements, which are used as a means of indicating a potential loss of control of licensee radiation protection program and trigger a requirement for specific action to be taken if exceeded. For additional information on ALARA, dose limits, and action levels, please see Attachment 8.
84	Netherlands	15	p.79	Are the NPPs using a value of unit collective dose (dollars/mSv) to decide on which ALARA-measures to take or not? Are these values recommended by the government of Canada?	Currently the Canadian Nuclear Safety Commission does not recommend specific dollar values for a unit of collective dose saved, it is left to the licensee's discretion to set this value. Some nuclear power plants use a value of unit collective dose (dollars/mSv) to decide on which measures to take with respect to the “as low as reasonably achievable” (ALARA) principle. For example: Ontario Power Generation, in N-STD-RA-0018, states that facility management and ALARA committees should consider a value of $25,000 per person-rem (~$2.5 million per person-Sievert) averted when evaluating cost-benefit options for dose reductions. New Brunswick Power Nuclear states a value of $4000/mSv in IR-03400-04, “ALARA Dose Reduction Five-Year Plan”. While Bruce Power recognizes there is a cost associated with dose, it has not assigned a specific monetary value given that the cost may differ depending on the circumstances.
85	Pakistan	15	Article 15.c Page 85	The data provided by the National Dose Registry in the table given in Annex-15 c, presents the average annual worker	The dose in the National Dose Registry (that is, the data provided in the table in Annex 15.c) includes all workers that are monitored under a facility's dosimetry program. The dose reported is the dose received for work performed at that particular facility, including contractor dose. This data does not include dose(s) received by a contractor from work performed at another facility within that time period.

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				dose, the collective dose and the maximum worker dose at Canadian NPPs for the period of 2001–2005. Does this data include the dose to the contractor's workers? If not, then how the exposure to casual or contractor workers is counted.	
86	Switzerland	15	Page 80	Radiological Exposure Permits. On which basis do the ALARA sections give radiological exposure permits? Do they use checklists to approve RP-planning?	Job dose targets are assessed for adherence to the "as low as reasonably achievable" (ALARA) principle and entered into radiological exposure permits. Checklists are not used. However, operating experience is considered when planning a job and radiation protection issues are discussed before performing the work. For additional information on ALARA, dose limits, and action levels, please see Attachment 8.
87	Switzerland	15	Page 80	Airborne Tritium Reduction. Which measures are taken into account to reduce incorporation of tritium? At which tritium concentration levels do the NPPs start measures?	Direct and indirect measures are used to reduce intake of tritium. Direct measures include tritium reduction facilities, leak reduction programs, and ventilation systems and driers. Indirect measures include training workers on tritium hazards and the use of personal protective equipment (PPE). Furthermore, avoiding or minimizing exposure time, and selecting the appropriate PPE are taken into account in work planning to maintain doses as low as reasonably achievable (ALARA). Each nuclear power plant has different levels or conditions requiring the use of protective equipment. For instance, a filtered or isolating mask must be worn at very low levels of tritium, whereas plastic suits are used for higher levels of tritium. These levels are specified in the licensee's radiation protection program. For additional information on ALARA, dose limits, and action levels, please see Attachment 8.
88	Switzerland	15	Page 80	Source Term Reduction Programme. Do the radiation dose targets have to be	The Canadian Nuclear Safety Commission does not require licensees to provide radiation dose targets in advance. Such information is often provided during update meetings on, or verified during inspection of licensee radiation protection programs. No regulatory action would be taken when a radiation dose target is missed. However, as a component of their

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				reported to the CNSC in advance? Which actions will be taken when missing such a target?	"as low as reasonably achievable" (ALARA) program, the licensee would conduct post-work ALARA reviews to determine how much dose was received; why the targets were missed; and possible areas for improvement. For additional information on ALARA, dose limits, and action levels, please see Attachment 8.
ARTICLE 16: EMERGENCY PREPAREDNESS					
2A	Pakistan	16.1	Section 16, Page 83	Kindly provide information on the classification of emergency conditions.	The shift supervisor initiates activation of the on-site emergency response organization, including the requirements for off-site notifications, situation updates and confirmations of any radioactive releases using the categorization/classification predefined in provincial emergency plans. The emergency classification system exists to promptly communicate accident severity and is initially based on the facility conditions. The shift supervisor is responsible for classifying an abnormal event, and the classification is re-evaluated when significant changes in station and radiological conditions occur. Depending on the nuclear power plant, classification of potential emergencies may vary. In general, however, three levels are used – alert, site emergency and general Emergency. An **alert** would be an event involving a localized hazard that can be confined and controlled by the on-site emergency response team. This level would include unknown events that warrant increased readiness or assessments. In the case of a **site emergency**, this could be an event resulting in a major decrease in the level of protection for the station personnel and an increased risk to the public requiring an enhanced degree of readiness by off-site authorities. A **general emergency** would be the result of an event with significant actual or potential radioactive release that may require the implementation of urgent protective actions for the public near the station and dose control for on-site personnel. This might include actual or potential core damage or measurement of doses off-site warranting urgent protective actions.
89	France	16.1	p. 83	Although the report is quite extensive, the presentation of this chapter does not provide	Implementation measures for severe accident management guidelines and an example of such implementation measures at Point Lepreau are detailed in Attachment 9.

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				a clear idea of the means and measures which would be implemented in practice in case of a severe accident. Could Canada develop these aspects in its presentation to the review meeting?	
90	Luxembourg	16.1	Page 84, resp 179-182	The Canadian provinces and territories are responsible for off-site emergency planning. Assuming that slightly different approaches are adopted in each province, Luxembourg is interested to hear which differences exist between the provinces in terms of zone radii around the NPP's, intervention levels for countermeasures and the organization of iodine prophylaxis. How does this affect an emergency situation?	Canada has three provinces with nuclear power plants: Ontario, Québec and New Brunswick. These provinces are responsible for establishing and adopting emergency planning zones that fit their needs, taking into account factors such as the number of facility units/capacity and population groups around the facility. Usually the licensee, the provincial emergency management organization and the Canadian Nuclear Safety Commission work out the details together based on the worst-case scenario. For example, for nuclear power plants in Ontario, a 10-km primary zone and a 50-km secondary zone have been adopted. Quebec has an 8-km primary zone and a 70-km secondary zone. In New Brunswick, the primary zone is 20 km, whereas the secondary zone is necessary to satisfy the geographical location of the power plant. The provinces and territories have different protective action levels based on radiation hazards and resources available. Radiation hazards and protective action levels for off-site emergency planning are described in Attachment 10.
91	Pakistan	16.1	Article 16.1, Page 85	It is described that there are 19 federal departments/agencies involved in the FNEP. What are the prime responsibilities according to Canadian legislation of CNSC during 'off-site' emergencies and the regulatory role in the Federal Nuclear	Health Canada is the custodian of the Federal Nuclear Emergency Plan (FNEP). The Canadian Nuclear Safety Commission (CNSC) is a supporting member of the FNEP and has specific duties to perform under this plan. For example, the CNSC sends technical, communications, coordination and executive representation to the FNEP's Emergency Operations Centre to assist with emergency response. Some of the functions include radiation protection advice, International Nuclear Event Scale coordination, radiation dispersion assessment, and coordination with the licensee and government authorities. Another important aspect is the media coverage for provincial and federal authorities to ensure harmonization of news bulletins and communiqués released to the public. During an on-site emergency, CNSC inspectors have a role to perform in the areas of

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				Emergency Plan (FNEP). Moreover, in case of on-site emergency, what are the functions of CNSC in the licensee's emergency plan?	communication, assessment, support and advice, and are linked to CNSC headquarters through emergency lines and computer electronic mail. They join the licensee's management team in their Emergency Operations Centre and perform the following activities: • monitor the situation and licensee's actions; • evaluate, with headquarters, the quality of licensee actions; • ensure needed information on the incident and on recovery actions is relayed to CNSC headquarters; • ensure that questions from the CNSC are considered by licensees; • recommend further actions from the CNSC; • ensure that any comments or questions from other government departments and from the licensee or requests for help or advice, are relayed to the appropriate management levels; • request for actions or deliver Orders if needed; and • in general, ensure that regulatory requirements (for example, the provisions of the *Nuclear Safety and Control Act*, the applicable regulations and licence conditions) are being complied with by the licensee. Emergency exercises are held at each site as per licence condition, and the CNSC is involved and participates regularly. CNSC site inspectors may also be requested to join the Provincial Operational Centre.
92	Ukraine	16.1	Annex 16.1 b, page 174	How CNSC interacts with Bruce NPP, emergency centres in Kincardine municipal area and Ontario region with the purpose of coordinated information of mass-media and public at the national level?	The Canadian Nuclear Safety Commission (CNSC) has resident staff at the Bruce site who will assemble at the licensee's site management centre when it is activated. Findings will be reported to the CNSC headquarters in Ottawa. The CNSC will send representation to Kincardine to act as a media/public relations spokesperson to answer questions and to assist the licensee and the province (municipality) with their response. In addition to sending staff to Kincardine, the CNSC will also deploy staff to the Provincial Emergency Operations Centre in Toronto to ensure the response received from the operator is well understood and provide advice to key organizations and stakeholders, including the media.
ARTICLE 17: SITING					
3A	Pakistan	17.1	Section 17, Page 90	Please indicate the magnitude of Safe Shutdown Earthquake (SSE) for different sites as per evaluation using HAD102/02.	The Canadian approach defines the design basis earthquake (DBE), which is equivalent to the safety shutdown earthquake, as being linked to a probability of exceedance of 1 in 1,000 years. Due to the lack of seismic standards for nuclear power plants at the time of construction, the "early-built" CANDU plants (Pickering A and Bruce A), were designed against the

#	Country	CNS Article	Report Reference	Question	Answer
					requirements of the National Building Code of Canada. Furthermore, the static analysis method was used for their design. Later on, they were re-assessed and retrofitted against review level earthquake values of 0.235 g for Pickering A and 0.15 g for Bruce A. Subsequent CANDU plants followed the rules of the Canadian Standards Association's CSA N289 series. The following DBE values are used: 0.05 g for Pickering B and Bruce B, 0.08 g for Darlington, 0.2 g for Point Lepreau, and 0.15 for Gentilly-2.
93	Slovenia	17.1	Art. 17(ii) b, p.92	Discovery of new fault lines affecting seismicity at the site; What happens in case they are found?	The probability of such a fault being seismogenic could be determined using a variety of tools. If that probability is non-zero, the fault would be considered as a contributing source and added to the existing hazard model. Then, to recalculate the seismic hazard, many characteristics of the fault would need to be discovered and estimated, such as the recurrence intervals of earthquakes on the fault and the magnitude of slip. The addition of the fault might or might not appreciably change the estimated hazard, depending on the relative contribution of the fault compared to the other sources that had already been included. It is likely that even in well-mapped places, not all active faults have been found. The best protection from a sudden rise in the estimated seismic hazard is a thoroughly researched seismic hazard model, together with a realistic assessment of its uncertainties. If the estimate of the seismic hazard has not been made thoroughly, it is possible that the newly discovered fault would contribute to an increase in that estimate.
94	Slovenia	17.1	Art. 17 (ii) b, p. 92	Changes to man-made neighbouring facilities such as a newly constructed oil refinery, rail corridor, airport flight path or chemical plant; Is there a consent required prior to building a facility with potential impact on NPP?	There are no land use planning guidelines that explicitly require consideration of the effects of changes to existing neighbouring man-made facilities on a nuclear power plant (NPP). Where such a proposed facility could potentially impact the safety analysis of the existing NPP, the owner/operator of the NPP is obligated under the *Nuclear Safety and Control Act* to conduct a review of the NPP safety analysis and notify the Canadian Nuclear Safety Commission of any changes that would affect safe operation of the plant under their licence.
95	Switzerland	Article 17.3	Page 92	Does the periodic verification of the continued acceptability of the NPP also include a re-evaluation of the site-specific seismic hazard and, if yes, is a probabilistic seismic	The site- specific seismic hazard is monitored with the use of seismic instrumentation at the federal level (Geological Survey of Canada), provincial level and the site level at each nuclear power plant. Continuous monitoring and recent scientific data did not warrant any changes to any original site-specific design basis earthquake. The seismic analysis of systems, structures and components (SSCs) is reassessed only when modifications are made to a seismically qualified SSC, or in response to research findings, analysis findings or operating experience. For example, the probabilistic seismic hazard analysis has been used for the development and acceptance of the review level earthquake for the Pickering A

#	Country	CNS Article	Report Reference	Question	Answer
				hazard analysis (PSHA) part of the re-evaluation?	seismic margin assessment. This station did not have a site-specific seismic hazard defined at the time of its design and construction.
ARTICLE 18: DESIGN AND CONSTRUCTION					
96	France	18.1		Could Canada describe CANDU designer organisation to draw lessons of operating experience feedback from reactors operating in foreign countries (mainly India and Korea…), beside IRS system?	The CANDU Owners Group (COG) provides an information exchange program to enhance excellence in the safety, reliability and economic performance of CANDU plants worldwide by sharing operating experience (OPEX). A weekly COG OPEX screening meeting teleconference, chaired by COG, serves as a CANDU screening committee to review event reports from CANDU stations and nuclear industry sources for applicability and significance to CANDU units. The screening committee consists of OPEX staff from Ontario Power Generation (Darlington, Pickering and Head Office), Bruce Power, Gentilly-2, Point. Lepreau, Cernavoda, Embalse, Wolsong, Qinshan Phase III, Atomic Energy of Canada Limited (AECL)'s Sheridan Park and Chalk River, the World Association of Nuclear Operators (WANO) and COG. Each site presents information about recent events at its location, which they believe may relevant to the other sites. COG presents nuclear industry reports, which it screens from sources such as WANO, IAEA and the United States Nuclear Regulatory Commission. OPEX feedback from Indian and Pakistani reactors comes through WANO participation in the COG OPEX screening meeting, as well as IAEA reports screened by COG. AECL has a feedback program for the systematic collection, recording, evaluation, and distribution of design, construction, commissioning and operating feedback information to AECL groups for the purpose of improving the safety and efficiency of the AECL product. Nuclear industry reports obtained through COG are collected, reviewed, and distributed for further evaluation.
3B	Argentina	18.1	p. 95	Concerning the deterioration of the containment concrete structures with age, please detail CNSC position about this issue, mainly in relation to the admitted leakage rate and the test required to control it.	In Canada, the Canadian Nuclear Safety Commission Regulatory Document R-7 and the Canadian Standards Association N287 govern the design, construction, commissioning and in-service inspection of the concrete containment structures (CCSs). One of the licensing conditions for nuclear power plant licensees is to develop and implement an in-service periodic inspection program for the CCSs. Licensees are required to perform periodic in-service inspection and testing of the CCSs at specified intervals to ensure their structural integrity and leak-tightness are maintained. The licensees submit the inspection and testing results and their evaluations to the regulator for review. If inspection results indicate an adverse trend, the regulator may require the licensee to increase the frequency of the inspection and/or provide compensatory measures.
97	Pakistan	18.1	Article 18 (i) b, Page 95	Reference Section 18(i) b page 95 - '-Barriers to Radioactive releases".	In Canada, the Canadian Nuclear Safety Commission (CNSC) Regulatory Document R-7 and the Canadian Standards Association Standard N287 govern the design, construction, commissioning and in-service inspection of the concrete containment structures (CCSs).

#	Country	CNS Article	Report Reference	Question	Answer
				The containment concrete structure which is the fourth barrier to prevent radioactive releases, may deteriorate with age. How is the aging of this structure analyzed by CNSC and what are the measures taken to address this issue. Is the ageing effects are incorporated in PSA?	One of the licensing conditions for nuclear power plant licensees is to develop and implement an in-service periodic inspection program for the CCSs. Licensees are required to perform periodic in-service inspection and testing of the CCSs at specified intervals to ensure their structural integrity and leak-tightness are maintained. The licensees submit the inspection and testing results and their evaluations to the regulator for review. If inspection results indicate an adverse trend, the regulator may require the licensee to increase the frequency of the inspection and/or provide compensatory measures. Currently, probabilistic safety assessments are not capable of predicting ageing effects or plant risk profiles, because the model is static in time. In 2006, the CNCS initiated a research project to address this issue.
98	Finland	18.2		Have you met specific problems to find spare parts or replacement components properly qualified to a high safety class, as needed for plant lifetime management? If yes, how have you addressed the problem?	The Canadian industry has taken a number of initiatives to deal with this issue. One is a joint procurement agreement through the CANDU Owners Group (COG). This agreement allows the industry to procure a number of replacement parts through COG for the mutual benefit of the industry, by creating enough demand for manufacturers to produce the required parts. The Canadian industry also has developed some capability to reverse-engineer and manufacture replacement parts that are no longer available. Canadian utilities also have quality assurance requirements embedded in their licences, which require a review of the vendors' quality programs to ensure a quality product is delivered and is qualified to the proper level.
ARTICLE 19: OPERATION					
99	Euratom	Article 19.2	p. 104, section 19 (ii) b	Was the information and experience gained with the implementation of the "Safe Operating Envelope Project" disseminated to other countries operating CANDU type reactors?	At the CANDU Senior Regulators meetings held in 2001 to 2003, participants were briefed on the status of the Canadian licensees' safe operating envelope projects. All CANDU-owning countries also attended a joint IAEA/Canadian Nuclear Safety Commission/CANDU Owners Group (COG) meeting on the safe operating envelope in 2001. Any significant findings (there have been none to date) would also have been reported to other CANDU operators via the COG operating experience exchange system.
100	United Kingdom	19.2	Page 104	The recognition that safe operating limits [SOE] needed to be better defined so that they are "…readily measurable by operations staff" is much	The safe operating envelope (SOE) projects were not established to address systematic shortcomings in licensees' compliance with operating limits. Various SOE projects were established by licensees, and not the Canadian Nuclear Safety Commission (CNSC), to better document the link between safety analysis and operational documentation. Licensees had identified shortcomings in the transparency and maintainability of the previous methods, and the project was aimed primarily at addressing these shortcomings. Licensees

#	Country	CNS Article	Report Reference	Question	Answer
				welcomed, as is the reported progress of the licensees on their SOE projects. Given the previous situation, it seems surprising that it was initially possible to properly specify the design requirements for safety-significant systems (see Section B2, 10, on page 15), let alone for CNSC staff to monitor compliance with the safe operating limits. One assumes that these should have formed part of the limits described in the Operating Policies and Principles, OP&Ps (see Section 19 (iii) on page 105). Could CNSC say more about how CNSC staff previously ensured compliance with the SOE limits? Is it now planned to amend the OP&Ps to reflect the clarified SOE limits?	and the CNSC have never expressed a generic concern that limits are not established, documented or adhered to. Early practice generally limited operational documentation to specific parameter limits relevant to normal operating conditions. Operating experience showed that this made the process of ensuring the station always operated within the “analysed envelope” quite labour intensive when equipment needed to be taken out of service. This was because limits were often expressed in a generic and highly conservative manner, which required configuration specific interpretation and development of temporary documentation changes. The SOE projects are mainly aimed at improving the usability of operating documentation. Ensuring that safety limits are readily measurable by operations staff can be as simple as expressing a limit as a tank level rather than as a liquid volume, after allowing for uncertainties such as tank geometry and orientation, location and calibration of level measurement and indication accuracy. The CNSC monitors compliance with limits through regulatory standard S-99. This standard requires that licensees report when a limit defined in licensing documentation is exceeded (in 6.3.1 (3)), or may be invalid (in 6.3.2.3 (b)). Licensees are also required to report when defined specifications for a special safety system or safety-related system become invalid (6.3.2.3 (d)). It should be noted that the SOE projects have identified no significant shortcomings in the previous safety system settings. Changes to the current licensing and compliance documentation have been discussed between the CNSC and licensees. There are currently no firm plans to make changes, though this remains a possibility.
101	Bulgaria	19.3		Do you have long term operation strategy or plans to operate the NPPs beyond design lifetime.	The current industry strategy for operation of CANDU plants beyond the design lifetime includes a refurbishment outage where some major components are replaced and safety upgrades are made to the plant. These refurbishments would add 25 to 30 years of additional plant life. The Canadian Nuclear Safety Commission (CNSC)’s expectations for life extension of nuclear power plants are given in RD-360, which is available on the CNSC Web site at www.nuclearsafety.gc.ca.

#	Country	CNS Article	Report Reference	Question	Answer
102	Bulgaria	19.3		What criteria are used to determine the lifetime of the plant.	All of the plants in Canada are of the CANDU design. The lifetime of CANDU reactors are typically limited by lifetime of the pressure tubes. Pressure tubes are nominally designed for a 30 year life.
103	Bulgaria	19.3		Do you have a re-qualification program for components to be used beyond their design lifetime.	There currently is no re-qualification program for components being used beyond their design lifetime. Current methods to determine component suitability for continued use are based on fitness-for-service guidelines and applicable codes and standards. Analysis of inspection results are compared with applicable safety factor derived acceptance criteria from the applicable codes/standards to determine if the component is fit for service beyond its design lifetime.
104	Korea, Republic of	19.3	Section 19(iii)	(Article 19-3, Section 19(iii)) CNSC's regulatory standard "Reliability programs for nuclear power plants (S-98 Rev.1)" specifies the identification of systems important to safety. The way of identification includes PSA, deterministic analyses and expert panels. - Please provide the procedure and methodology for the identification in detail. - Please provide the actual application case. - Do you have any cases that the PSA results were used for optimizing maintenance or test interval?	The process used to identify systems important to safety is documented in the procedure described in CANDU Owners Group (COG) guideline *Interim Implementation Guidelines for CANDU Nuclear Plant Reliability Programs* (COG-05-9011), prepared under the COG Industry Risk and Reliability Technical Working Group. The process is generally followed in two steps: identification of systems important to safety based on two risk measures (“Risk Achievement Worth” and “Fussell-Vesely”); and the subsequent use of an expert panel to finalize the list by adding systems if necessary. Expert Panel consideration includes uncertainty; completeness; accident analysis in the safety report; regulatory specifications; relevant past assessments; and impact of operating in shutdown state. Probabilistic safety assessment and system reliability analyses have been used to risk-inform changes recommended for test and maintenance intervals by preventative maintenance optimization program.
105	India	19.4	Page 108, section 19 (iv)	In the context of follow up of Pickering blackout incident of August 14, 2003, it is reported that modifications have been	The modifications entailed removing the transfer from automatic to manual voltage control, as well as limiting the power system stabilizer output to +/- 6% for units 5 to 8.

#	Country	CNS Article	Report Reference	Question	Answer
				completed to the T/G controls for continuation of unit operation during a similar event. Please indicate the modifications carried out.	
106	Switzerland	19.4	Pages 106 – 109	Do any of the procedures mentioned need a formal approval by the regulatory body? Does the overall structure of the procedures need a formal approval by the regulatory body?	The procedures mentioned under Section 19 (iv) do not require the approval of the Canadian Nuclear Safety Commission (CNSC). However, for the Darlington Nuclear Power Plant (NPP), the procedures requiring formal regulatory approval are given in section 4 of regulatory documents R-7, R-8 and R-9 (available on the CNSC Web site at www.nuclearsafety.gc.ca). These documents were developed in the late 1970s and apply only to reactors licensed for construction after January 1, 1981. As a result, the requirements they contain apply only to the Darlington nuclear power plant (NPP). Many of the procedures requiring regulatory approval under R-7, 8 and 9 are contained in the Operating Policies and Principles (OP&Ps) for each NPP in Canada. Because the OP&Ps are formally approved by the regulatory body, these procedures are therefore also approved through this process. In general, the majority of the procedures requiring formal regulatory approval are those for taking corrective actions in the event that a special safety system is found to be impaired when required to be available, and those for intentionally taking a special safety system out of service. The overall structure of the procedures does not require formal approval by the regulatory body.
107	Finland	19.7		Please explain your national policy and practice of sending feedback reports to the international interested parties on actions that have been taken in your country as response to significant events reported through international channels (e.g., WANO, IRS).	Actions taken in Canada in response to events reported internationally are presented annually by the Canadian Nuclear Safety Commission through its delegates to the appropriate fora such as the IAEA Incident Reporting System technical committee and the OECD-Nuclear Energy Agency Working Group on Operating Experience.
108	Finland	19.7		Please explain how the regulatory body ensures	All NPPs in Canada have a licence condition that requires the licensee to meet the requirements of Canadian Standard CSA N286.5. Section 6.4 of this standard requires the

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				or verifies that the operators are informed and properly analyse the operating experiences reported through the well established international channels (e.g., WANO, IRS), and that they address the lessons learned by taking proper actions.	licensee to have an operating experience (OPEX) program. At each Canadian power plant, analysis of operating experience and associated activities forms part of the OPEX program, which is subject to routine inspection and compliance activities by the regulator. The OPEX programs incorporate both national and international events.
109	Finland	19.7		Please explain the principles or criteria applied by the regulator and operator for screening other experience than incidents (e.g., management issues, unexpected degradation, design weaknesses, external hazards not considered earlier), for the purpose of ensuring adequate sharing of important experience with international interested parties (regulatory bodies, operators, de-signers, international bodies). Identify the relevant guide documents, if any, used for the screening.	Issues arising from experience, other than events, are reported in different fora within the regulatory body such as management meetings and inspection reports. Screening of these issues to share with the public and international fora is performed as part of the preparation of the Significant Development Reports, which are submitted to the Commission members. A guide is now under development to aid in the determination of issues to share with the public and international fora. At nuclear power plants, significance determination is made by following specific operational procedures. For example, at Ontario Power Generation (OPG), the significance of events other than “incidents” (unexpected degradation of equipment, management issues raised through various means including World Association of Nuclear Operators (WANO) peer reviews, design weaknesses) is rated using criteria in the Corrective Action Program governance [N-PROC-RA-0022]. This significance is then used in the Operating Experience (OPEX) program (governance document N-PROC-RA-0035) to determine the potential impact on other stations within OPG, the Canadian nuclear industry, or the larger international community (via the Institute of Nuclear Power Operators/WANO/IAEA channels).
110	France	19.7	General	As reported in the third Canadian report, follow-up to the loss of the electricity grid (blackout) on August 14, 2003, at	The baseline probabilistic safety assessment (PSA) for Pickering B does include the event sequence involving a loss of off-site (class IV) power, referred to as the “loss of bulk electricity system” (LOBES) event, and a simultaneous random failure of on-site (class IV) power resulting in the unavailability of the high-pressure emergency coolant injection system pumps. In the case of the 2003 LOBES, the failure of the on-site (class IV) power

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				Pickering identified that some of the design and operation assumptions could be challenged by such an event. In particular, the high-pressure ECC system, which is common to both Pickering A and B, was unavailable for 5.5 hours because of loss of power to the high-pressure pumps. In addition, the emergency high-pressure service water system restoration for all Pickering B units was delayed because of low suction pressure supplying the emergency high-pressure service water pumps. Did Canada quantify the core damage frequency increase of this event? More generally, does Canada use Accident Sequence Precursor program to assess the potential impact of safety significant events?	was attributable to system deficiencies that were identified and corrected. With these deficiencies corrected and with an integrated and ongoing evaluation of unit survivability, Ontario Power Generation believes the reliability of the on-site (class IV) power system meets the assumptions in the PSA. Hence, the baseline core damage frequency (CDF) estimates are unchanged. In terms of characterizing the CDF impact of the LOBES event itself, the relatively short duration of the event coupled with provisions to provide adequate core cooling through other means, results in a relatively small increase to public risk. However, in view of potential adverse conditions (that could have resulted had additional components failed), the Pickering B plant response to the LOBES was classified as a level 2 event (incident) on the International Nuclear Event Scale.
111	France	19.7	p. 109 – 19(vii)	Could Canada indicate if, in addition to the experience feedback analysis presented in the report, a probabilistic analysis of experience	If the operating experience relates to a component failure mode or failure frequency, then a probabilistic safety assessment (PSA) analysis may be used to assess the impact and the PSA assumptions and fault trees may be updated as appropriate. If the operating experience is not equipment based, then it is unlikely that its impact would be analyzed using PSA. The decision of whether to use the PSA would be made using the expert judgment of individuals evaluating the operating experience.

#	Country	CNS Article	Report Reference	Question	Answer
				feedback (precursor program) is carried out?	
112	Romania	19.7		In Appendix D (page 121) is presented an event which caused the failure in the process of transferring fuel from the reactor. We would like to know more about component failure, direct cause and root cause.	The direct cause of the event at Gentilly-2 was a fuel-clad failure during unloading of channel P-15 combined with an out-of-balance ventilation system, which resulted in a radiological contamination of accessible rooms in the reactor building. A few days before this event, the concentration of Xe-133 in the coolant started to slightly increase, but the clad failure location system did not provide significant results. A fuel-clad failure while unloading channel P-15 was therefore not suspected. The ventilation system was out of balance because of a combination of factors: the installation of a PFU (portable fan unit) that pushes air in suction lines; a possible change by a worker of the draft adjustment of a PFU without any control process in place; a damper wrongly positioned; and possibly another damper repositioned without having performed tests with a smoke pencil. The root cause of this event was inappropriate management of temporary changes made to the reactor building ventilation system, from installation to removal of ducts and PFUs.
113	Romania	19.7		In Appendix D (page 121) is presented the loss of regulation event which happened at Bruce A Unit 3. We would like to know more about component failure, direct cause and root cause.	Bruce A Unit 3 was operating at full power when the helium pressure of the liquid zone control system (LZCS) increased due to a fault in the pressure control circuit, which caused the feed valve to go to the fully open position. At the time of the fault, the backup pressure controller was set to manual to facilitate the control of an elevated hydrogen level in the system. The rapid increase in helium pressure caused the water in the liquid zones to drain, resulting in a reactor power increase. The reactor power increase was sensed by the reactor regulating system (RRS), which automatically started to compensate by adding water to the liquid zones. However, the rate with which the RRS could add water to the liquid zones could not compensate for the drain rate. A second reactivity control system (stepback) also sensed the increase in power and was activated in preparation of terminating the reactor power increase. Both of these responses from the RRS functioned as per their design. The reactor power increase was also sensed by both shutdown systems that activated, and the reactor was automatically shut down. Within one minute of the initial failure in the helium pressure control circuit, the fault was cleared and the LZCS returned to normal operation. Activities completed after the event were focused on confirming that station systems had responded as designed; understanding the initiating fault; and preventing a repeat event. The controllers were refurbished and dynamically tested. Operating procedures were also reviewed and revised and a design change was implemented to avoid a similar event.

#	Country	CNS Article	Report Reference	Question	Answer
					The root cause was attributed to inadequate staff adherence to new procedures for controlling hydrogen. Staff had continued to use the old method due to the following factors:, operating instructions were not consistent in relation to conditions under which the LZCS could be set to manual mode; and an additional complication was caused by operational difficulties in using the approved method of hydrogen control. For additional and detailed information, please see Attachment 11.
114	Russian Federation	19.7		It is not quite clear from Annex 19 (vii), Programs to Collect and Analyse Information on Operating Experience, who is responsible for assessing operating experience feedback effectiveness and how this assessment is performed. Do the operating organization and regulatory body assess the operating experience feedback effectiveness? Who, in particular, performs this assessment and in what way?	At each Canadian nuclear power plant, analysis of operating experience and associated activities form part of the operating experience (OPEX) program. The Canadian Nuclear Safety Commission (CNSC) routinely assesses the OPEX program through inspection and compliance activities. As well, licensees conduct their own assessments through internal audits and self-evaluation programs. Periodic reviews (most often using the utility's internal self-assessment process) of the effectiveness and impact of the entire OPEX program would be conducted approximately every two years by the program owner. More frequent, but more narrowly focused self assessments, looking only at particular portions of the program where metrics and corrective action program feedback have indicated potential weakness, may be conducted at the discretion of the program owner or the direction of senior management. The CNSC conducts periodic assessments of the utility's OPEX program effectiveness, usually in concert with a wider review including the Corrective Action Program, and at a frequency determined by the CNSC. On site, the CNSC inspectors examine each OPEX process (including data collection, reporting, report screening, root cause analyses and the identification and disposition of corrective actions) to ensure that lessons learned have been transmitted to the appropriate groups within the utility. In addition, CNSC specialists perform systematic desktop review of every reported incident to ensure licensee compliance with the regulatory requirements and safety performance standards. Furthermore, the CNSC dispatches focussed inspection teams to perform independent assessments of certain high-profile incidents.
115	Switzerland	19.7	Page 109	Does the authority conduct a quality surveillance of the licencees operating experience analysing programme?	At each Canadian nuclear power plant, analysis of operating experience and associated activities form part of the operating experience (OPEX) program. The assessment of the OPEX program is subject to routine inspection and compliance activities by the regulator. As well, the licensees, through their internal audits and self-evaluation programs conduct their own assessments.
116	China	19.8	CH‡W 19	Does Canada have	A) Canada does not have specific criteria for categorization of low-level waste. Currently,

#	Country	CNS Article	Report Reference	Question	Answer
				specific criteria for categorization of very low radioactive waste produced in nuclear power plant? What are the treatment methods for very low radioactive waste?	radioactive wastes in Canada are classified into three categories based on origin and radiological hazard: nuclear fuel waste, low-level waste and uranium mine and mill tailings. Individual licensees are free to utilize more detailed classification systems for their own waste management programs. There is an initiative by the Canadian nuclear industry, including nuclear power plant operators, to incorporate a classification system into a Canadian Standards Association (CSA) document (CSA N292.3). Canadian Nuclear Safety Commission (CNSC) staff are participating in the development of CSA N292.3 rather than developing a CNSC regulatory guide. The CSA document is based on the IAEA draft Safety Guide DS-390,*Classification of Radioactive Waste*. The proposed radioactive waste classification system includes a very low-level radioactive waste category. The document is currently in draft form and is to be published in the next reporting period. B) There are currently no treatment methods for very low level wastes at a Canadian nuclear power plants (NPPs). As mentioned above, this classification does not currently exist in Canadian NPP procedures. Current practice classifies material as either clean (meets clearance and free release limits) or low-level waste. The creation and issue of CSA N292.3 will assist in the development of treatment options for very low-level waste in Canada. For additional information on radioactive waste management, please see the *Canadian National Report for the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management*, available on the CNSC Web site at www.nuclearsafety.gc.ca.
117	Switzerland	19.8	Page 110	Does the RAW management use a clearance level for the exemption from regulatory control?	Exemption from licensing requirements is currently addressed in the *Nuclear Substances and Radiation Devices Regulations* but there is currently no explicit provision for clearance levels in the regulations. However, licensees may apply to the Canadian Nuclear Safety Commission (CNSC) on a case-by-case basis to obtain regulatory approval of clearance activities where there is no unreasonable risk to the heath, safety of persons and the environment. A project to amend these regulations is currently underway, which will include a provision for clearance levels. These proposed amendments will better align Canada's approach for exemption and clearance of radioactive material from regulatory control. The amended regulations will consider the IAEA Basis Safety Standards as well as the most recent guidance from the IAEA on the concepts of exemption, exclusion and clearance. (IAEA-RS-G-1.7) CNSC staff expects to have these amendments approved by the Governor in

#	Country	CNS Article	Report Reference	Question	Answer
					Council and publication in the Part II of the *Canada Gazette* in the next reporting period.

Attachment 1: Excerpts From the *Canadian Environmental Assessment Act* – Section 37 and Related Material

37. (1) Subject to subsections (1.1) to (1.3), the responsible authority shall take one of the following courses of action in respect of a project after taking into consideration the report submitted by a mediator or a review panel or, in the case of a project referred back to the responsible authority pursuant to subsection 23(1), the comprehensive study report:

> ***23.*** *(1) The Minister shall, after taking into consideration the comprehensive study report and any comments filed pursuant to subsection 22(2), refer the project back to the responsible authority for action under section 37 and issue an environmental assessment decision statement that*
>
> (a*) sets out the Minister's opinion as to whether, taking into account the implementation of any mitigation measures that the Minister considers appropriate, the project is or is not likely to cause significant adverse environmental effects; and*
>
> (b*) sets out any mitigation measures or follow-up program that the Minister considers appropriate, after having taken into account the views of the responsible authorities and other federal authorities concerning the measures and program.*
>
> **22.** (2) Prior to the deadline set out in the notice published by the Agency, any person may file comments with the Agency relating to the conclusions and recommendations and any other aspect of the comprehensive study report.

(*a*) where, taking into account the implementation of any mitigation measures that the responsible authority considers appropriate,

(i) the project is not likely to cause significant adverse environmental effects, or

(ii) the project is likely to cause significant adverse environmental effects that can be justified in the circumstances,

the responsible authority may exercise any power or perform any duty or function that would permit the project to be carried out in whole or in part; or

(*b*) where, taking into account the implementation of any mitigation measures that the responsible authority considers appropriate, the project is likely to cause significant adverse environmental effects that cannot be justified in the circumstances, the responsible authority shall not exercise any power or perform any duty or function conferred on it by or under any Act of Parliament that would permit the project to be carried out in whole or in part.

(1.1) Where a report is submitted by a mediator or review panel,

(*a*) the responsible authority shall take into consideration the report and, with the approval of the Governor in Council, respond to the report;

(*b*) the Governor in Council may, for the purpose of giving the approval referred to in paragraph (*a*), require the mediator or review panel to clarify any of the recommendations set out in the report; and

(*c*) the responsible authority shall take a course of action under subsection (1) that is in conformity with the approval of the Governor in Council referred to in paragraph (*a*).

(1.3) Where a project is referred back to a responsible authority under subsection 23(1) and the Minister issues an environmental assessment decision statement to the effect that the project is likely to cause significant adverse environmental effects, no course of action may be taken by the responsible authority under subsection (1) without the approval of the Governor in Council.

Attachment 2: Development and Use of the Risk-Informed Decision-Making Process

In 2005, a Risk Management Working Group was formed to advance the development and implementation of a risk-management process into the Canadian Nuclear Safety Commission (CNSC) regulatory framework. The group was chartered to develop outputs that should enhance the CNSC's capacity with respect to:

a) regulation of the development, production, possession or use of nuclear power reactors in order to prevent unreasonable risk to the environment and to the health, safety and security of persons;
b) assessment of health, safety, security and environmental risks associated with potential problems, and the use of risk management principles to set priorities for regulation and regulatory changes, such that the limited resources available to the CNSC are used where they do the most good;
c) planning of regulatory activities based on an analysis of relevant risks, the results of previous regulatory activities, and a rigorous, well-documented process linking activities to required results, in addition to the judgment and expertise of staff; and
d) implementation of a quantifiable rating of safety performance, taking into account the safety-related portion of systems used in the industry, and use of this rating, along with a more rigorous and integrated risk assessment and other qualitative information, to systematically determine the level and type of regulatory effort required.

The group's specific tasks included:

a) preparing an internal document that defines "risk management" in the CNSC's regulatory context, describes the basic concepts of risk and risk management, highlights typical risk decision-making situations at the CNSC, and outlines a decision-making process for managing risk;
b) organizing orientation and discussion workshops for CNSC power reactor program staff and management on risk management concepts;
c) identifying appropriate risk management tools and methods, including qualitative and quantitative methods, and the practical application of the endorsed approach to different situations;
d) communicating with stakeholders; and
e) following up to assess integration in the power reactor regulatory program.

The group yielded a seven-step risk informed decision making (RIDM) process, which is primarily based on Canadian Standards Association standard *Risk Management: Guideline for Decision-Makers* (CSA-Q850-97). The process will be carried out by a team of CNSC staff who collectively possess the necessary knowledge of the issue being considered and the surrounding circumstances. The team will makes its recommendations to the CNSC decision-maker who had initiated the process. The process calls for continual documentation, as well as consultation with stakeholders throughout.

The RIDM process has been through successful trial use since May 2006 and has subsequently been used satisfactorily in numerous applications in the power reactor regulatory program. For example, the RIDM process was used for the following functions: ranking outstanding safety issues (including the Generic Action Items) with consideration of the likelihood and consequences of scenarios where such issues may be of importance; focusing research efforts on safety-significant areas; facilitating development of plant-specific safety improvement programs or review of new reactor designs; and directing staff's effort by strategically focusing on important areas.

Lessons learned from the field trial and the other uses suggest, among other things, the importance of the following:

- delivering adequate training on the use of the process;
- making proper team selection;
- correctly identifying and agreeing on the issue at hand;
- ensuring that data and information used are accurate and current; and
- conducting adequate consultation with stakeholders.

The Risk Management Working Group is in the process of recommending to the CNSC management that the RIDM process be declared “operational” and be formally incorporated, or referenced, in the CNSC Management System Manual.

Attachment 3: Transparency of the Decision-Making Process of the Canadian Nuclear Safety Commission

The Canadian Nuclear Safety Commission (CNSC) can be best described as the watchdog governing the use of nuclear energy as well as nuclear substances and devices in Canada. It is one of only a few nuclear regulators in the world that involves the public in the conduct of its hearings and meetings.

The Commission Tribunal (usually referred to simply as the Commission) is an independent quasi-judicial administrative tribunal consisting of up to seven Commission Members appointed by the Governor in Council (Canadian federal government). The Commission takes into account the views, concerns and opinions of interested parties and intervenors when establishing regulatory policies and making licensing decisions. For licensing matters, CNSC staff prepares recommendations for Members of the Commission, who make the final decisions after hearing from interested parties (the applicant and public intervenors) via the public hearing process. Public hearings are a highly visible component of the work of the Commission, which holds approximately 30 public hearings each year, aggregated in about 20 hearing days. Matters heard in the context of public hearings are those involving nuclear power plants, uranium mines and mills, nuclear waste facilities and research reactors. Other licensing activities — for example, those related to nuclear substances and devices, as well as import and export — have been delegated by the tribunal to CNSC staff.

The NSCA requires that the Commission hold public hearings for most licensing matters that come before it for decision. The NSCA also allows the Commission to hold public hearings on any other matter within its jurisdiction if the Commission determines it is in the public interest to do so. This is in addition to the meetings of the Commission, which are also generally open to the public. Note that in-camera or closed sessions may be held on sensitive issues, such as security matters. The NSCA requires that before the Commission makes a licensing decision, it must give the applicant or licensee an "opportunity to be heard". In the interest of fairness, the Commission gives the person most affected by the decision the opportunity to present their views to it before making its decision. A public hearing is structured so as to give affected parties and in most cases, interested members of the public, a reasonable opportunity to make submissions — in writing and/or orally — in relation to the matter to be decided by the Commission.

The CNSC *Rules of Procedure* facilitate and encourage active participation by members of the public. In addition to notifying the applicant or licensee, the Commission gives 60 days' advance notice of a public hearing in a manner that is likely to come to the attention of interested members of the public. As a general rule, the notice of public hearing is posted on the CNSC Web site and is also published in newspapers serving the area in which the facility is located. The notice supplies information on the duration of the hearing (one or two days), its purpose, dates, time, place and the deadlines for filing documents prior to the hearing. Participants may attend in person to make their presentations or have their written submissions considered in a public forum. Members of the public may also attend and observe the proceedings without further formality. In order to participate actively in the hearing, interested persons must seek and be granted the status of an intervenor by the Commission. Public hearings are usually well attended by members of the public and of the media, and may include a number of intervenors (for example, individuals, unions, employees, community and environmental groups). The Commission has a public hearing room in Ottawa, but may periodically conduct hearings at different locations across the country to provide a greater opportunity for the public to participate in or observe its proceedings. The Commission has been using, where appropriate, teleconferencing and videoconferencing in the conduct of proceedings, and plans to continue its move toward a greater use of available technologies. For example, the Commission is now video webcasting some of its proceedings where matters have significant public interest. In addition, transparency is assured through the issuance of detailed Records of Proceedings, including Reasons for Decision, so that the decisions of the Commission reflect the evidence submitted and the rationale of the Commission for the decisions.

Public participation in Commission proceedings has ensured that the views of persons interested in nuclear energy facilities are heard and factored into the decisions of the Commission. Public proceedings have also served to increase the effectiveness, visibility and credibility of the Commission. Transparency of the licensing process is a cornerstone of the CNSC regulatory framework.

Attachment 4: Rating of Safety Areas and Programs

“Program” and “implementation” ratings for each safety area are derived through a process whereby CNSC specialists, site inspection supervisors and/or Regulatory Program Division (RPD) Officers evaluate several review topics in each safety area and program, and present their findings to the regulatory program directors for discussion and approval. The process is outlined in an internal work-instructions document that is updated annually. There is currently no prescribed algorithm for rolling up the evaluations of the review topics, such as using an average, weighted average, or lowest score methodology. Reviewers instead seek to balance the rating of safety performance in terms of stated performance objectives in each of the safety areas and programs.

The following describes a number of improvement initiatives and the tools, resources, and the process that the CNSC to assign an overall rating for the safety performance of Canada’s nuclear power plants (NPPs).

Improvement Initiatives

1. Performance objectives have been established for each program and safety area;
2. Internal work instructions have been developed and circulated for field use;
3. Rating forms were developed to standardize the review and assessment of each safety area and program;
4. Surveys (2005, 2006) were circulated to staff to collect feedback and establish lessons learned; and
5. A working group was formed to analyze survey results and produce recommendations for improvements for both the report development process and the report itself.

Tools

1. Table A3.14.4.2 on page 89 of Canada’s Third National Report, showing safety areas, programs and review topics;
2. Program rating forms that contain systematic analysis of selected review topics, using expert judgement and/or identified performance measures against findings from compliance activities (S-99 reports, inspections, events analysis, corrective action follow-up);
3. Safety area rating forms that contain information transferred from program rating forms as well as the roll-up rating for the safety area; and
4. Internal work instructions.

Resources

1. CNSC inspection reports;
2. Licensee reports submitted in accordance with Regulatory Standard S-99, *Reporting Requirements for Operating Nuclear Power Plants*;
3. Other licensee documents, as appropriate; and
4. Relevant communications (internal and external).

Rating Process

1. Review of programs:
    - Programs are assessed by CNSC specialists, site inspection supervisors, and/or RPD officers;
    - Resources are used to gather information and data relevant to the program (Table A3.14.4.2 may be used as guidance);
    - Rating forms are completed based on analysis of, and conclusions reached using information gathered;
    - Each program rating form includes a summary review for each review topic, and an overall grade for the “program” and “implementation” along with an overall assessment for both “program” and “implementation”;
    - If more than one staff member is involved in the review of a program, a consensus is reached in order to complete the rating form;

- The assessment expressed in the rating forms are agreed by the division directors of all contributors to the review of the program;
- The rating form is reviewed and approved by the RPD director responsible for the subject NPP; and
- In case of unresolved disagreement, the final decision is made by the RPD director.

2. Review of safety areas:
   - Safety area rating forms are completed based on expert judgement and communication with peers, using the grades and summary reviews in the program rating forms;
   - Each completed safety area rating form includes a single rating for "program" and "implementation" and a summary assessment for each;
   - The ratings expressed in the rating form are agreed to by the division director of the specialist or the RPD officer involved;
   - The rating form is reviewed and approved by the RPD director responsible for the subject NPP;
   - In case of unresolved disagreement, the final decision is made by the RPD director; and
   - The following conflict resolution mechanism is considered:
     - The RPD director is responsible for the final decision regarding the rating of programs and safety areas; and
     - In making this decision, the RPD director:
       - takes into consideration other factors, information, and overall state of the subject NPP;
       - endeavours to reach a consensus with all involved staff members; and
       - provides justification for the decision made.

3. Overall CNSC staff assessment of the Canadian nuclear power industry's safety performance

   In the *Annual CNSC Staff Report on the Safety Performance of the Canadian Nuclear Power Industry*:
   - Grades or ratings from rating forms are tabulated along with descriptive summaries of the analyses and justifications;
   - A summary assessment of an NPP performance is included if the Industry Report coincides with the mid-term report for that NPP; and
   - An overall assessment of the Canadian nuclear power industry's safety performance is included as to the degree to which CNSC expectations were met in the safety areas.

Attachment 5: Safety Performance Indicators System Used by Canadian NPPs

The safety performance indicators systems used internally by the nuclear power plants (NPPs) are based on measures devised by managers to determine their ability to meet their departments' current safety performance objectives and targets. Each department has devised its own statement of purpose, which describes how it supports the stated NPP objectives and priorities.

As an example, at New Brunswick Power Nuclear (NBPN), specific departmental objectives and targets are established for each department's operational plan, based on the balanced score-card approach.

NBPN currently has three overall station objectives:

1) safe and reliable operations;
2) execution of refurbishment on time and on budget; and
3) achieving world-class performance.

These objectives require a significant focus on safety. Key refurbishment milestones can only be achieved if safety and quality requirements are met. As an example, the Nuclear Safety Unit has, as part of its statement of purpose,

- Define the safe operating envelope and verify that the station is operated in a safe manner consistent with claims made in support of the reactor operating licence;
- Support the resolution of issues related to safety analysis; and
- Obtain and maintain licences and address safety questions raised by the regulator.

Some of the performance indicators devised to determine the Nuclear Safety Unit's ability to meet these objectives are:

- Number of station work advanced planning meetings missed;
- Number of safety reviews of proposed station work not completed by advanced target date;
- Progress on projects to resolve safety assessment and analysis issues compared to schedule; and
- Progress on issues to address refurbishment regulatory commitments and regulatory requirements for fuel re-loading and restart following refurbishment.

The departmental manager sets specific targets for each performance indicator, and each measure has a green, yellow or red rating established against these targets:

Green - meeting or ahead of targets
Yellow - at risk of not meeting targets
Red - not meeting targets.

Charts displaying the status of each measure are updated regularly and displayed prominently within the department. The balanced score-card approach requires appropriate measures for all departmental objectives, but it also requires targets to be balanced such that an appropriate mix of green, yellow and red is achieved, so that areas requiring resource allocation changes are identified early enough to remedy adverse trends. All green charts are taken as a sign of low achievement targets; all red charts are taken as a sign of unrealistic targets and project schedules. Given that much work at the station involves components and contributions from multiple departments and work units, overall success depends on a good balance when negotiating and committing to schedules.

It is a constant challenge to devise performance indicators that are comprehensive enough to ensure work units and departments are aligned to, and capable of, achieving departmental objectives in support of overall station objectives, and that targets are sufficiently challenging, but realistic.

Attachment 6: CNSC Human Factors Regulatory Program and Staff Competency

CNSC Human Factors Regulatory Program

The CNSC human factors review areas include human factors in design, human reliability analysis, work organization and job design, procedures and job aids, human performance, performance monitoring, performance improvement and organization and management.

Assessment of the human factors review areas involves many different types of regulatory activities. Licensee programs are evaluated during licensing or re-licensing actions. CNSC staff then conducts compliance inspections of licensee programs in the different review areas to assess the adequacy of the documented programs and the effectiveness of their implementation. Human factors specialists also review documents submitted by the licensee in response to regulatory requirements or as requested by the CNSC. This provides a further opportunity to assess the adequacy of licensee programs in these areas. Licensees are required to report events which meet certain regulatory criteria. CNSC staff analyzes these events to identify emerging safety issues and trends.

The results of these regulatory activities are integrated by human factors specialists and used to determine the overall effectiveness of the licensees' programs and their implementation. Human Factors specialists at the CNSC use internal regulatory documents as well as accepted international standards in the evaluation and rating of licensee programs and their implementation. Human factors ratings are included within the Performance Assurance safety area, along with Quality Management, Training and Personnel Certification. Ratings are published in the "Annual CNSC Staff Report on the Safety Performance of the Canadian Nuclear Power Industry" which is published on the CNSC Web site at www.nuclearsafety.gc.ca.

CNSC Human Factors Staff Competency

At the CNSC, senior human factors specialists are expected to possess at least a master's degree in human factors engineering, industrial engineering, engineering psychology, ergonomics or other related degree. Most of the senior human factors specialists (there are seven) at the CNSC possess a Ph.D. in human factors. In addition, it is considered desirable for such specialists to have in-depth relevant experience in the development and implementation of human factors programs in a process industry environment (for example, nuclear, aviation, chemical, transportation), preferably a high-reliability industry. Senior human factors specialists must also be eligible for membership in a recognized Human Factors professional society, such as the Human Factors and Ergonomics Society (USA), the Human Factors Association of Canada, or the Ergonomics Society (UK).

The knowledge that a senior human factors specialist is expected to possess includes the following:

- sound knowledge of human factors principles, theories, methods, standards, and guidelines;
- sound knowledge of the human factors issues that are applicable to the life-cycle of nuclear generating stations, including human error, human-system interface design, work organization, job design and procedures design;
- extensive knowledge of human cognitive and physical capabilities and limitations, including memory, attention, information processing, decision- making and anthropometrics;
- good knowledge of human factors methods and techniques including function analysis, task analysis, human error analysis, workload analysis, verification and validation methods; and
- good knowledge of human-machine and human-computer interface design and assessment issues that are applicable to nuclear facilities in Canada, including the areas of information display, alarm/annunciation, decision support systems, control design, workplace layout and workstation design.

The senior human factors specialists, along with an organization and management specialist reside in the Human and Organizational Performance Division of the CNSC. Generally speaking, oversight activities specifically related to the discipline of human factors (for example, design, training, procedures, hours of work, etc.) are carried out by the senior human factors specialists. Where issues related to management or safety culture comes into play, the organization and management specialist would also be involved.

Attachment 7: Major Enhancements to Shutdown System Capability and Improvements to the Emergency Core Coolant System at Pickering A

1. Engineering Changes Completed to Enhance Shutdown System Capabilities

The following changes were carried out to enhance the shutdown system capabilities of Pickering A units 1 and 4.

A shutdown system enhancement (SDSE) was added to enhance the existing Pickering A shutdown system A (SDSA), to further reduce the probability of failure to shutdown. To the extent practicable, the SDSA and SDSE were made independent of each other (subject to constraints resulting from a retrofit of new equipment into an operating plant), from trip sensing to the final relays in the shutoff rod drop logic and the moderator dump logic (does not include shutoff rod clutch mechanisms or moderator dump valves). The enhancement provided a new set of triplicated trip sensors and trip logic augmented with new moderator dump logic. The SDSE trip parameters are neutron overpower (NOP), high log rate (HLR), heat transport high pressure (HTHP), heat transport low pressure (HTLP) and manual trip. The enhancement also included the addition of two shutoff rods, bringing the total number to 23. The equipment was selected and installed to meet environmental qualification requirements.

The SDSE provides a means of detecting conditions requiring reactor shutdown, in addition to the SDSA system. The SDSE independently initiates automatic operation of the existing reactivity control devices (for example, shutoff rods and/or moderator dump valves) following a Large Break Loss of Coolant Accident (LLOCA) or Fast Rate Loss of Reactor Power Control (FLORPC) and provides automatic trip coverage for a spatial LORPC. It also initiates a moderator dump if the reactor power rundown is inadequate following a detection of a trip condition by the SDSE. It also ensures that the addition of SDSE moderator dump arrest units does not significantly affect the reliability of moderator as a heat sink following a LLOCA with shutdown by the shutoff rod system. The two new shutoff rods and SDSE heat transport high- and low-pressure trip parameters enhanced shutdown depth of the shutoff rods and the trip coverage for process failures, which do not result in fast power transients.

2. Engineering Changes Completed to Improve the Emergency Core Cooling System reliability

For Pickering A, the modifications completed in the Emergency Coolant Injection system (ECI) are divided into two groups:
Group 1: To reduce the predicted severe core damage frequency
Group 2: To perform system upgrade to meet environmental qualification, seismic requirements and other system improvements

Group 1: Reduce the current predicted severe core damage frequency to below the OPG Corporate Probabilistic Risk Assessment (Ref: NA44-CORR-00531-00085).

As background, moderator/ECI recovery failures represent the second-largest combined risk reduction worth. The current crosslink between moderator cooling and ECI recovery exposes ECI recovery to numerous process-related failures over the post-LOCA mission used in the Pickering A risk assessment. By eliminating the leading failure modes associated with EC1 recovery the frequency of severe core damage was reduced to 5×10^{-5} (events/yr.).

The following design modifications have been completed to the moderator/ECI recovery system:

(a) Reconfiguration of tempering flow path to allow closure of calandria outlet valves (COVs) and eliminate associated process control failures;
(b) Reconfiguration and resizing of dump tank outlet valves to provide a reliable alternate source of tempering flow;

(c) Change of failure positions of COVs and dump tank outlet valves to match long term ECI Recovery fail-safe positions;
(d) Provision of backup cooling to moderator heat exchangers;
(e) Modification of vault recovery valve opening logic;
(f) Separation of moderator room air conditioning units' 48Vdc control power supplies; and
(g) Provision of calandria outlet tie line for COV testing.

Group 2: System upgrades to ECI, moderator, heat transport and associated systems (major modifications only) comprised the following:

(a) Redesign and replacement of ECI recovery strainers;
(b) Replacement and upgrade of shutdown cooling valve actuators, stems and wedges;
(c) Duplication of D_2O recovery valves;
(d) Replacement and upgrade of ECI recovery injection valves, actuators, drain valve and associated instrumentation;
(e) Replacement and upgrade of high-pressure injection valve actuators and limit switches;
(f) Replacement and upgrade of calandria and dump tank level transmitter and associated instrumentation (in-core LOCA detection instrumentation);
(g) Replacement and upgrade of ECI primary heat transport (PHT) pressure transmitter and associated instrumentation;
(h) Replacement of ECI boiler room pressure transmitter and associated instrumentation;
(i) Replacement and upgrade of moderator room active drainage pump motor; redesign of sump level instrumentation and pump discharge piping;
(j) Replacement and upgrade of both moderator room air conditioning units and reconfiguration of cooling ducts;
(k) Replacement and upgrade of moderator calandria inlet and calandria outlet valves, actuator and associated instrumentation;
(l) Replacement and upgrade of pneumatic vault recovery valves and electric vault recovery valve actuators;
(m) Removal and blank of calandria vault sump outlet valve;
(n) Replacement and upgrade of both moderator heat exchangers (used for ECI recovery);
(o) Replacement and upgrade of five moderator pump motors (used for ECI recovery);
(p) Provision of back-up instrument air supply to ECI equipment;
(q) Seismic upgrade to ECI equipment motor control centres
(r) ECI EQ cable replacement;
(s) Upgrade of uninhabitable main control room and ECI instrumentation;
(t) Provision of new cooling heat exchanger for ECI storage tank; and
(u) Replacement of PHT boiler room piping insulation (ECI loss of coolant accident debris issue).

Attachment 8: ALARA, Dose Limits, and Action Levels

ALARA

Section 4 of the Canadian Nuclear Safety Commission (CNSC) *Radiation Protection Regulations* requires that every licensee shall implement a radiation protection program and shall, as part of that program, keep the amount of exposure to radon progeny and the effective dose and equivalent dose received by and committed to persons as low as is reasonably achievable (ALARA), social and economic factors being taken into account. This is achieved through the implementation of the following:

(i) management control over work practices;
(ii) personnel qualification and training;
(iii) control of occupational and public exposure to radiation; and
(iv) planning for unusual situations.

Dose Limits

In addition, Section 13 of the CNSC *Radiation Protection Regulations* requires that every licensee shall ensure that the following effective dose limits are not exceeded: 50 mSv in a year and 100 mSv over 5 years for a nuclear energy worker; 4 mSv for a pregnant nuclear energy worker for the balance of pregnancy; and 1 mSv for a person who is not a nuclear energy worker (public).

Action Levels

Paragraph 3(1) f of the CNSC *General Nuclear Safety and Control Regulations* requires that an application for a licence shall contain any proposed action levels. An "action level" is defined in Subsection 6(1) of the CNSC *Radiation Protection Regulations* as a specific dose of radiation or other parameter that, if reached, may indicate a loss of control of part of a licensee's radiation protection program and triggers a requirement for specific action to be taken. Subsection 6(2) of the same Regulations requires that when an action level is reached, the licensee must report to the CNSC, conduct an investigation to establish the cause for reaching the action level, and identify and take action to restore the effectiveness of the radiation protection program. Paragraph 6.3.2.1 b of the CNSC S-99, *Reporting Requirements for Operating Nuclear Power Plants*, requires the licensee to file its report to the CNSC within 45 days of being aware that the action level was reached. In 2001, the CNSC issued a regulatory guide G-228, *Developing and Using Action Levels*, for use by applicants for a CNSC licence, other than a licence to abandon.

Attachment 9: Implementation Measures of Severe Accident Management Guidelines in Canada

The implementation of Severe Accident Management Guidelines (SAMG) at nuclear power plants (NPPs) in Canada is in various stages of completion at the different facilities. The measures to be implemented would differ somewhat depending on the location and nature of the NPP, as some are single-unit facilities in relatively remote rural locations and others are multi-unit facilities close to major urban centres. Off-site emergency response measures are the responsibility of the provincial government in which the NPP is located and differ in the measures that would be implemented. In general, off-site emergency response plans are comprehensive enough to encompass the management of severe accident response; exercise scenarios have often tended to focus on the more severe events in order to require testing of the full scope of provincial plans. Most changes are anticipated in relation to facilitating implementation of on-site severe accident management by the NPP owners, the licensed operators of the facilities.

The Point Lepreau NPP in New Brunswick is currently implementing SAMG and undertaking some revisions to the related emergency plans. As an example of the measures that would be implemented for a severe accident, we can use this NPP, which is a single-unit facility owned by a small electrical utility in a relatively remote rural location, about 1 hour and 40 minutes travel time by road from provincial capital of Fredericton. Because of the relatively remote location, the provincial government took the decision when the NPP was first constructed to pre-distribute iodine tablets to all homes around the NPP site. In the event of a severe accident at Point Lepreau, the provincial government nuclear emergency response plan would be activated, and a decision would be made based on reported NPP status, current weather conditions and wind direction, and the status of evacuation routes, on the best measures to protect the public. If conditions suggested the best response would be evacuation, a pre-selected evacuation route and destination would be chosen, reception centres activated and residents within 20 km of the NPP would be automatically notified of the incident, the recommended evacuation action, destination and route by the Community Notification Service. This service sends notification and a selected message to a visual display in residents' homes. The service also reports by exception which residents have not been sent the message, and these residents are individually visited by community wardens. The service can alternatively recommend shelter indoors and taking the pre-distributed iodine tablets, in the event that the provincial emergency response organization determines that this would be a better option.

The NPP operations response to a severe accident would be to declare a general emergency at the NPP site and activate the on-site emergency response. Changes to facilitate severe accident management have focussed on the incident command organization at the facility, with the emphasis on providing the essential safety and mitigation functions of “control, cool and contain” by any capable and available means. At Point Lepreau, changes are being made to the on-site incident command structure to more closely align with the standard Canadian all-hazards approach to emergency management and to facilitate interaction and coordination with other emergency response agencies. As part of these changes, it is planned to relieve the on-duty shift operations crew at the earliest opportunity and to turn over on-site emergency management to a more comprehensive emergency management crew under the leadership of an incident commander. This approach recognizes more comprehensively that the nature of the command function has changed significantly, and that the incident command location, the size and focus of the response crews, the equipment and resources to be used, the locations from which it might be operated and the processes and guidance to be considered are potentially significantly different in the event of a major traumatizing incident such as a severe accident.

Attachment 10: Radiation Hazards and Protective Action Levels in Off-Site Emergency Planning

A short-term hazard may be external radiation resulting from a radioactive plume; for example, inhaled radioiodine would be absorbed by the thyroid gland. Some types of accidents could lead to hazards from radioactive material deposited on the body, ground and food. The long-term hazard in many cases is likely to be from the consumption of contaminated foods, especially milk and water.

Typically, provincial emergency plans provide protective actions levels (PALs), which are countermeasures to protect the population living near the NPP. Also, when considering the application of PALs to a particular sector either in the primary or secondary zone, certain groups within the general population may need special consideration. For example, pregnant women — and in some cases, children or infants — are considered in this vulnerable group, which would also include patients in hospitals and institutions, bedridden residents in nursing homes, handicapped persons and prison inmates.

PALs aid in planning and decision-making during an emergency. Expressed in terms of projected radiation doses, they provide technical guidance on the need to take specific protective measures, such as evacuation, sheltering, limiting access to the affected zone(s), and thyroid blocking. PALs for the banning of consumption of affected foods and water are expressed as levels or radionuclide concentrations.

When the time is available for making decisions is limited, it would be entirely appropriate to use PALs as the only technical criteria for indicating the need for the application of any protective measure. However, when the urgency does not exist and when dealing with low doses over a long period, if is preferable to consider, in addition to the PALs, other technical factors such as collective dose and its likely health impact.

Since thyroid blocking protects the body only against radioiodine, and the latter may be just one component of a hazardous radioactive emission, thyroid blocking should be considered and used in conjunction with other protective measures.

Where other protective measures can provide effective protection against all or most of the components of hazardous radioactivity, including radioiodine, they are to be preferred over thyroid blocking. This applies to evacuation in the case of the hazard from inhaled radioiodine, and other available measures against the hazard from ingestion or radioiodine. The option of undertaking thyroid blocking should be available to those persons who may be unable to evacuate at the same time as other people, such as the sick and infirm people, essential services, emergency workers, detainees, etc.

The Canadian Nuclear Safety Commission is cognizant of these approaches and can adjust accordingly to address the emergency in question.

Attachment 11: Sequence of Events that Resulted in Loss of Regulation at Bruce A Unit 3

Bruce A Unit 3 was operating at full power when the liquid zone control system (LZCS) helium pressure increased due to a fault in the pressure control circuit, which caused the feed valve to go to the fully open position.

At the time of the fault, the backup pressure controller was set to the manual mode to facilitate control of an elevated hydrogen level in the system. The backup pressure controller is designed to automatically compensate for increased helium pressure, but as the controller was set to manual, it could not respond to the increased pressure. The pressure therefore rose rapidly, causing water in the liquid zones to drain. This resulted in a reactor power increase.

The reactor power increase was sensed by the reactor regulating system (RRS), which automatically started to compensate to reduce the reactor power, by adding water to the liquid zones. However, the rate with which the RRS could add water to the liquid zones could not compensate for the drain rate caused by the increased helium pressure and reactor power increased. A second reactivity control system (stepback) also sensed the increase in power and one of the two arming circuits was activated in preparation to terminate the reactor power increase. Both of the above responses from the RRS functioned as per design.

The reactor power increase was also sensed by both shutdown systems (1 and 2), and the power level reached their trip setpoints, prior to stepback, terminating the power increase. Both shutdown systems activated and the reactor was automatically shutdown.

Within one minute of the initial failure in the helium pressure control circuit, the fault cleared and the LZCS returned to normal operation.

<u>Components</u>

The PICs in this event are Fischer-Porter Model #53EL4431BDKBAA, 350-550 KPAD controllers.

The apparent cause of the transient was a failure on the 3-63480-PT62/PIC62 control loop, which resulted in feed valve 3-34810-CV79 failing open and 3-34810-CV87 failing closed, while 3-34810-CV86 was set at 10% open, with 3-63480-PIC63 on manual. The LZCS operated as expected, given the failure and the control lineup at the time of the transient. The 3-63481-PIC63 control loop was in manual mode for increased recombination flow and did not respond to mitigate the event.

Loop connections on 3-63480-PT62 were checked for proper connection and/or corrosion, per on-line wiring. Several loose connections were discovered on PT62 associated wiring. A break in any of these connections would have resulted in a loss of signal to PIC62 and the failure defined above. All loose connections were tightened, and no other anomalies were found. The fault was identified as an intermittent fault in the pressure transmitter PT62, which, in turn, sent an inaccurate signal to the pressure controller.

The following actions were taken:

- 3-63480-PIC62 was replaced. Initial inspection of the removed PIC62 did not reveal any faults with the controller.
- 3-63480-PT62 45 VDC power supply common loads were checked for faults coinciding with the transient. None were found.
- A modification was made to the system to provide a manual means of increasing recombiner flow to maintain system chemistry within specification.
- A data logger was installed on 3-63480-PT62 wiring to monitor the input signal during unit operation.

Actions completed following the event

The activities following the event focused on three areas:
1. Confirmation that the station systems responded as designed;
2. Full understanding of the initiating fault; and
3. Prevention of a repeat event.

Assurance that the station equipment had performed as per design was accomplished through a detailed review of the response of the reactivity control systems (RRS, SDS1, SDS2). This confirmed that all systems functioned as per their design and that the power increase had no detrimental affect on the fuel.

Assurance that the fault was understood and removed was accomplished by rigorous testing of the helium pressure control circuits, on both the normal and backup circuits. The controllers were refurbished, and dynamic testing of the control loops was completed. To provide further assurance, additional monitoring equipment was connected to the control circuit to collect data should another fault occur after the unit had been returned to normal operation.

To ensure that the LZCS backup helium controllers would not be set to manual operation, the following activities were completed:
1. A review of all associated operating procedures was performed to confirm that the manual mode of operation was not permitted;
2. All controllers which have the capability of being placed into manual mode operation were inspected to confirm that they were selected to the correct mode of operation; and
3. A design change to the helium circuit was implemented to improve the hydrogen recombination capability and so avoid the need for operating staff to manually control the hydrogen levels.

Root cause analysis

An experienced multidisciplinary team completed a root cause analysis of the event and attributed it to staff’s less than adequate adherence to procedures.

Specifically, the LZCS operating instructions had been revised to prevent the controllers being set to manual mode when controlling hydrogen concentrations. However, not all the operating instructions were consistent: elsewhere in the LZCS operating instructions, the selection of manual mode for the LZCS controller was permitted, and the chemistry control procedure required the controller to be set to manual mode to control hydrogen concentration. An additional complication was caused by the operational difficulties in using the approved method of hydrogen control.

Each of the above factors contributed to the operating staff continued use of the old method of controlling hydrogen and failure to adhere to the correct procedure.

To address this weakness, further enhancements of the field observation and coaching (FO&C) program have been implemented. The objectives of the FO&C program are to observe staff executing their work and to provide immediate feedback on their performance in meeting defined expectations and standards. The enhancements include paired observations designed to ensure that expectations of the VP Operations are known, understood and being practised by all operations staff. The paired observations, together with immediate reinforcement and coaching, will continue until the appropriate behaviours are consistently observed and the events caused by human performance errors are consistently low.

Contributing causes to the event were also identified in the following areas and are currently being remedied:
1. The monitoring of system health did not meet the requirements specified by the governing documents;
2. The risk management process was not being consistently applied and the risk of operating the controller in manual was not fully understood;

3. Had the design of the helium-purging arrangement been improved, the staff would have been more likely to use it and not place the controller on manual; and
4. The process to determine training needs from procedural revisions was inconsistent and did not always identify changes to long-standing practices.