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

Sixth Review Meeting
March 2014

Gouvernement du Canada
Government of Canada

Canada

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

This document is to accompany the Canadian National Report for the Convention on Nuclear Safety – Sixth Report.

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Canada’s Sixth National Report
for the Convention on Nuclear Safety

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This document supplements the Canadian National Report for the Sixth Review Meeting of the Convention on Nuclear Safety. By offering additional and detailed information in response to 176 specific questions, or comments received from 21 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 CNSC staff and representatives from Ontario Power Generation, Bruce Power, New Brunswick Power Nuclear, Hydro-Québec and Atomic Energy of Canada Limited.

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GENERAL COMMENTS					
1	Argentina	General	Chapter 1; Introduction; page 9	Regarding the decommissioning of Gentilly 2 and Pickering A Unit 2 and 3, could you please explain how it is expected to manage the heavy water after temporary storing? (Chapter 1; Introduction; page 9)	All of the heavy water that was removed from Pickering Units 2 and 3 as part of their safe storage following permanent removal from service was stored on site. The heavy water is being used for normal loss/make-up for the balance of the OPG nuclear fleet. G-2 is planning to sell part of its heavy water inventory and will be storing the remainder of its stock onsite.
2	Czech Republic	General	Page 8	You write that Advanced Fuel CANDU Reactor is designed to use alternative fuels such as recovered uranium from the reprocessing of used light-water reactor fuel, low-enriched uranium and plutonium-mixed oxide and thorium, in addition to the conventional natural uranium. The reason for using recovered uranium from the reprocessing of used light-water reactor fuel and plutonium-mixed oxide and thorium is obvious, but what would be the reason for using low-enriched uranium in CANDU reactor?	CANDU reactors in Canada have been, since the beginning, fuelled with natural uranium fuel. However, the use of heavy-water moderator and on-power refuelling makes the CANDU reactor feasible to burn alternative fuels such as recovered uranium from the reprocessing of used light-water reactor fuel, low-enriched uranium and plutonium-mixed oxide and thorium, in addition to the conventional natural uranium. With no exception, these options are being demonstrated with Advanced Fuel CANDU Reactor (AFCR) which is designed based on the Enhanced CANDU 6 (EC6). There are several reasons why one would consider using low-enriched uranium in CANDU-type reactors. First, use of low-enriched uranium to compensate the reactivity loss when the light-water is used as coolant (this is the case for the ACR-1000 design).

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					Second use of low-enriched uranium to compensate the reactivity loss when the poison is used in the fuel bundle to reduce positive coolant void reactivity (this is the case for both ACR-1000 design and the Low Void Reactivity Fuel design). Third, use of low-enriched uranium could lead to longer discharge burnup and hence better economy (this is the case for the ACR-1000 design). Finally, please note that the enrichment from recovered uranium from the reprocessing of used light-water reactor fuel is about 0.9 % which is considered as low-enriched uranium (the enrichment of natural uranium is about 0.7%).
3	Euratom	General	Summary, page 14	During the reporting period, the CNSC continued its progress in enhancing the regulatory framework – which included various regulatory documents relevant to NPPs (both existing NPPs and new-build projects) – along with progress in aligning the regulatory framework with international standards (as a minimum). These changes have been introduced into the regulatory framework in a risk-informed way. - When does Canada foresee that the process of aligning its regulatory framework with	The alignment of the CNSC regulatory documents with international standards is an ongoing process that continues with the regulatory framework modernization. Many of the current suite of CNSC regulatory documents and supporting national consensus standards have been informed by the international standards, and adapted for the technical specificities of CANDU technology, and the national legislative and statutory environment, and will continue to do so.

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				international standards will be complete? Will there be areas where Canadian standards will be preferred to international standards given the technical specificities of Canada's fleet?	
4	Euratom	General	Summary-challenge C-3, pages 16-17	The review of lessons learned from Fukushima concluded that SAMGs were generally adequate, but three specific actions were identified for follow-up: 1. Develop/finalize and fully implement SAMGs at each NPP. 2. Expand the scope of SAMGs to include multi-unit and irradiated fuel bay events (see the Canadian report for the Second Extraordinary Meeting for details). 3. Validate and/or refine SAMGs to demonstrate their adequacy to address lessons learned from Fukushima. Some licensees have completed the implementation and validation of SAMGs. Plans to address the additional enhancements to SAMGs are being developed. - What is Canada's timeline for	The deadline for completing work on these actions by all NPP licensees in Canada as established in the CNSC Integrated Action Plan on the lessons learned from the Fukushima Daiichi nuclear accident was December 31, 2013. Detailed submissions from licensees are currently under review and preliminary indications from Specialist evaluators are that licensees are on track to meet the objectives set-out in the Expectation and Closure Criteria for each action. Validation of SAMGs including human and organizational factors is presently scheduled for completion by December 2014. For example, Ontario Power Generation (OPG) has completed development and implementation of SAMGs, and has expanded these SAMGs to cover irradiated fuel bays. Work is still ongoing to complete the expansion of SAMGs to multi-unit events and to complete their validation Since Fukushima, OPG has carried out two separate sets of table-top exercises or drills at each of its facilities to

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				completion of the implementation and validation of SAMGs? - Have any significant unexpected difficulties emerged during the validation process to date?	validate and improve its SAMGs.
5	Germany	General	entire Report	The National Report of Canada is written in a legible, generally understandable and very well structured manner. Each article of the convention is comprehensively and thoroughly addressed giving the reader a clear picture of nuclear safety in Canada.	Comment is appreciated. Thank you.
6	Indonesia	General	p. 20/341 or p. 8	The NR mentions that Candu Energy acts as the original designer and vendor of the CANDU technology. Candu Energy has four reactor designs: • CANDU 6: The current fleet of reactors in operations is based on the existing CANDU 6 design, a heavy-water moderated reactor utilizing natural uranium fuel and on-power refueling; • Enhanced CANDU 6: A Generation III, 700 MWe class heavy-water moderated and cooled reactor that is based on the successful CANDU 6 model;	The CNSC does not certify designs. CANDU 6 reactors are licensed for operation in Canada (in Quebec and New Brunswick). The Enhanced CANDU 6, and ACR 1000 have been reviewed under a pre-project vendor design review. The vendor design review process is a proven and standardized process to evaluate, in principle, whether there are fundamental barriers to licensing the vendor's reactor design in Canada. Further information can be found on the CNSC's website, at http://www.nuclearsafety.gc.ca/eng/licenseesapplicants/powerplants/newapplicants/vendorpreproject/index.cfm

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				• Advanced CANDU Reactor (ACR-1000): An evolutionary, Generation III+, 1,200 MWe class heavy water reactor; • Advanced Fuel CANDU Reactor: Designed to use alternative fuels such as recovered uranium from the reprocessing of used light-water reactor fuel, low-enriched uranium and plutonium-mixed oxide and thorium, in addition to the conventional natural uranium. Could you provide more information whether the above reactor designs have been certified or not ?.	
7	Ireland	General	General	Ireland thanks Canada for providing a comprehensive report.	Comment is appreciated. Thank you.
8	Ireland	General	Ch II, p17 and Section 14(i)(d) p133-136	It is noted that the regulatory standard S-294 is being revised to consider multi-unit effects. Have CNSC and its partner emergency agencies (federal and provincial) considered applying the principle of 'extendibility' to emergency planning and protective zones around existing and future NPPs as a result of this revision?	The CNSC and partner federal organizations are currently involved in these discussions. Discussions are meant for all levels of government and the Operator. A Canadian Standard N-1600 is being developed to look at all aspects of emergency management. It should be released shortly. Ontario is one of the Provinces who intends to review their Emergency Plans and this would involve their emergency planning basis for severe accidents which are beyond design

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					basis. This could very well lead to making changes to the size of their emergency planning zones.
9	Norway	General	1	The Canadian report is very comprehensive and well structured.	Comment is appreciated. Thank you.
10	Romania	General	General	The report submitted by Canada is very comprehensive and detailed.	Comment is appreciated. Thank you.
11	Russian Federation	General	Section I, Subsection D.3	The Subsection describes plans of building 4 new reactors of maximum electric power capacity of 4,800 MW. What type of fuel is planned to use in new reactors?	The evaluation process for the addition of new nuclear power generation in the Province of Ontario considered both Pressurized Water Reactor (PWR) technology and Pressurized Heavy Water Reactor (PHWR) technology. Both technologies use uranium based fuel.
12	Switzerland	General	n.a.	Perusing the chapters of the Canadian national report, information on a protected facility to remotely shutdown the reactor, to operate and monitor the essential safety systems and parameters was not found. What are the Canadian regulatory requirements regarding emergency control rooms?	All Canadian facilities have secondary control rooms separated physically and electrically from the main control room. Requirements are found in RD-337 for new build.
13	Switzerland	General	Summary, p.18	Summary of other safety improvements during the reporting period In addition to addressing the three	Such events have long been considered in PSA (with appropriate supporting deterministic analysis) which comprises an integrated assessment of all accidents and events.

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				challenges from the Fifth Review Meeting and initiating numerous safety improvements in response to Fukushima, numerous other safety improvements were made at the Canadian NPPs during the reporting period. Were SBO / LOOP / SFP protection / Loss of heat sink / Combined events / Reevaluation of hazards integrated?	As part of the response to the Fukushima nuclear accident, Canadian licensees performed a generic assessment of the plant response to a long term loss of electrical power with consequential loss of heat sinks. The assessment included the response of the reactor and spent fuel pools. No specific cause was postulated. The response was judged to be bounding for the hazards arising from the combined events.
14	Switzerland	General	General	The last OSART missions to Canada have taken place in 2005. When will Canada host up to date OSART Missions again?	In light of Fukushima, and considering that nuclear safety has become more global post Fukushima, CNSC reiterated to Canadian licensees that OSART missions will be of great value and requested that operators commit to an OSART mission in 2015. Debate between operators is underway on whether an OSART mission can serve to replace a WANO mission.
15	Switzerland	General	n.a.	The Fukushima accident has shown that design and siting, especially the back-fitting of plants according to the state of the art of science and technology, are crucial topics. Is Canada in favor of extending the scope of OSART missions from mainly operational issues to	The CNSC Integrated Action Plan being implemented by Canadian licensees as a result of lessons learned from the Fukushima nuclear accident imposed a significant number of safety improvements, based on the latest state of the art science and technology, that are now being addressed through scheduled plant outages or planned refurbishment activities. The deadline for all Fukushima related improvements set-out in the Action Plan is

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				design and siting issues?	December 2015. This being said, the CNSC has no objections in principle to extending the scope of an OSART mission to include design and siting issues; given that the extra work is likely to result in tangible benefits.
16	Switzerland	General	D3, page 11 and 12	In the Canadian report it is written that there will be future construction of NPPs within the existing boundary of the Darlington site. How will the lessons learned of the Fukushima accident affect the construction of this new NPPs?	Any new plants at the Darlington site will have to meet regulatory requirements. CNSC regulatory documents, such as design, safety analysis, accident management, environmental protection and emergency preparedness, site suitability are being updated to reflect the lessons learned from the Fukushima nuclear accident.
17	Switzerland	General	D2, page 10	The Canadian report states that "Life extension is being pursued or considered for many of the reactor units at the Canadian NPPs." What were the consequences of the Fukushima accidents on the life extension plans for the Canadian NPPs? What were the major refurbishement issues due to the Fukhushima accident?	Most safety upgrades resulting from Fukushima lessons learned were already planned under refurbishment activities and are consistent with actions identified in the CNSC Integrated Action Plan in response to the Fukushima Daiichi nuclear accident. The Canadian approach to life extension of Nuclear Power Plants in Canada is based on one time application of a Periodic Safety Review (PSR), called an Integrated Safety Review (ISR). The ISR enables determination of reasonable and practical modifications that should be made to enhance the safety of the facility to a level approaching that of modern plants, and to allow for long-term operation. At the start of the reporting period for the 6th RM Canada had 20 operating units under three

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					different provincial authorities. Refurbishment activities at the Point Lepreau NPP in the Province of New Brunswick were completed in 2013 the station was returned to full service. Point Lepreau was the first Canadian NPP to complete major plant modifications set-out in the CNSC Integrated Action Plan on the lessons learned from the Fukushima Daiichi nuclear accident, before the Fukushima events (e.g., emergency Filtered Venting, Make-up to the Shield tank to prevent core-concrete interaction, installation of PARS). The Province of Ontario has for the time being indicated that it would not be initiating the construction of new NPPs but that refurbishment activities would continue on a number of units. All NPPs schedule for refurbishment will incorporate the upgrades identified in the Action Plan. The major refurbishment issues identified as a result of Fukushima were aimed primarily at providing additional emergency back-up power in case of prolonged station black-out and additional make-up water to critical heat sinks including the Spent Fuel Bays. Filtered venting and passive hydrogen mitigation systems have been or will be installed at all units before end calendar 2015. Additional emergency measures are being introduced by end calendar 2014 to ensure alternate off-site operating control centers are fully operational together with the installation of real-time boundary

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					monitoring stations at each site. All on-site and off-site emergency plans are to be tested during local, provincial and federal level exercises. In the Province of Quebec, for reasons unrelated to Fukushima, a decision was made not to refurbish its Gentilly-2 nuclear installation and the utility initiated decommission activities on termination of its operating licence in December 2012.
18	Switzerland	General	n.a.	Both Europe and countries in Latin America have undertaken Stress Tests at their NPPs. Why has Canada not undertaken Stress Tests similar to those in Europe in the light of the Fukushima accident?	The CNSC published its Fukushima Task Force Nuclear Power Plant Safety Review Criteria on its website in July 2011, which constituted the Canadian 'Stress Test'. In the process of formulating the safety review criteria, the CNSC Task Force considered all the applicable lessons learned from the Fukushima accident, and reviewed selected international reports to ensure that all aspects relevant to Canada were addressed. Effectively, the CNSC Task Force has subjected the Canadian nuclear power plants, the existing emergency response measures, and the regulatory framework and supporting processes to a systematic and comprehensive "stress test" to evaluate means to further protect the health and safety of Canadians and the environment. The CNSC Task Force monitored approaches taken by selected international task forces. The

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					terms of reference were reviewed to help validate the CNSC's approach. The selected international task forces were from the United States Nuclear Regulatory Commission (U.S. NRC) and the Western European Nuclear Regulators' Association (WENRA). The CNSC found that the three task forces had broadly similar terms of reference. The WENRA approach had provided the basis for many of the reviews that were performed around the world. WENRA developed its approach based on design and safety analysis techniques. This approach places particularly strong emphasis on giving a detailed and systematic evaluation of accident progression that considers successive failures of the mitigating measures and identifies key timings and potential cliff edges. WENRA explicitly mentions consideration of accidents at multiple reactors on a site but gives little emphasis to this aspect. The CNSC Task Force placed more emphasis on multi-unit events since the multi-unit reactors in Canada share parts of containment. The US NRC review was focused on the adequacy of its own specific regulatory requirements and did not include (at this stage) input from licensees. The European Nuclear Safety Regulators

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					Group (ENSREG) released its “stress test” specification following the CNSC review. Its specification was based closely on the WENRA stress test and did not change the CNSC Task Force review findings. The CNSC terms of reference and Nuclear Power Plant Safety Review Criteria are consistent in general terms with the approaches seen in the other international task forces and include the specific areas of emphasis identified above.

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ARTICLE 6: EXISTING NUCLEAR INSTALLATIONS					
19	France	Article 6	§ 6 (b) - p.20 Appendix B - p.205	The oldest NPP was constructed in 1969 and the most recent one in 1984. Does Canada intend to enhance safety by re-examining the former design assumptions through new studies, in order to bring the safety level of the older units up to the level of the recent ones?	The short answer is yes. The safety of Canadian NPPs is evaluated on a regular basis (every five years at the time of the licence renewal). One of the considerations for licence renewal is the level of compmpliand of the facility with modern codes and standards. As a general rule, compliance should be achieved to the extent practicale unless the licensee can show that undertaking the necessary changes will affect plant operation in such a way that overall safety is not enhanced. When NPP are being refurbished for life extension, Intetrated Safety Review guidelines require licensees to perform a comparison with modern codes and standards. Any gaps could result in upgrades in the design of the station. This would be summarized in the Integrated Implementation Plan resulting from the ISR. This exercise has been done at the Bruce A, Darlington, and Point Lepreau facilities.
20	India	Article 6	Page-20 & Appendix-B	Gentilly-2 is placed in a safe shutdown state since Dec. 12 to initiate decommissioning activities, whereas similar vintage units like Pt. Lepreau, Bruce-A units-1&2 have undergone life extension. Can Canada share	Decision as to whether to proceed with the refurbishment of an NPP rest with the responsible provincial authority. In the case of Gentilly-2, the increase in project costs, combined with the deterioration of the market price of electricity in North America, led Hydro-Québec to recommend to the

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				specific reasons for not extending life of Gentilly-2?	Government of Quebec the closure of the plant.
21	Russian Federation	Article 6	Article 6	Were Canada's reactors uprated? If yes, how was the uprating safety of the reactors justified? Did the uprating affect the level of risk of reactors (core damage frequency)?	No NPP licensee in Canada has applied for a reactor power uprate; however, turbine side uprates have been performed. A licensee wishing to pursue such a direction is required to apply to the CNSC for an amendment to the Operating Licence in the form of a Project Description. This request would trigger an Environmental Determination under the Canadian Environmental Assessment Act which would then lead to a decision on the type of Environmental Assessment that is needed for the project to proceed. Regardless of the route chosen, the applicant is required to demonstrate to the CNSC all the effects of this uprate on their existing safety case (through Safety Analysis) and demonstrate an adequate level of safety under the proposed new operating conditions. The new safety case would also need to consider aging effects over the remaining service life of the facility.
ARTICLE 7: LEGISLATIVE AND REGULATORY FRAMEWORK					
22	Spain	Article 7.1	pg. 47 and 284	According to the report, the life extension of NPP is carried out following the guidance provided by the Life Extension of Nuclear Power Plants (RD-360) that	It is not planned to replace the Pickering B pressure tubes as part of the limited life extension of the units. While the "assumed design life" of the pressure tubes is 30 yrs (210,000 Effective Full Power Hours (EFPH)),

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				require to perform an integrate safety review (ISR) and the corresponding integrated implementation plan. As a result, Bruce A and Point Lepreau NPP have decided to refurbish the plants in order to extend the life for approximately 25 years or more. In other way, Pickering B decided that incremental life extension until 2020, rather than the options of shutdown or refurbishment, was the best option In Annex 14 (i) it is written that the life-limiting components of the Pickering B unit are the pressure tubes and that a reassessment of those components predicts the end of their assumed design life in approximately 2015. How can the plant life be extended until 2020? Is it planned to replace those critical components?	or approximately end of calendar year 2015, extensive analysis and laboratory testing has been completed to demonstrate that the pressure tubes are fit for service for at least 247,000 EFPHs. The testing and analysis have been submitted to the regulator and the rationale for operation beyond the assumed design life has been accepted. In addition, a full ISR for the Pickering B units was completed, along with an Environment Assessment and a Global Assessment. The results of these assessments formed the basis for the Integrated Implementation Plan (IIP), which defined the inputs to the Continued Operations Plan (COP). The COP is the living document that describes all of the work that must be done to extend the life of station to at least 2020. Before the Pickering B station enters its life extension phase at the end of 2015, all of the COP actions will need to be completed.
23	Spain	Article 7.1	49	Under the new harmonize licenses, the licensees can implement design modifications in the plants as long as the modifications are within the licensing basis and executed according to the licensee’s management system. Could you provide some examples	The mechanism of written approval from the Commission has, in fact, not been used since CNSC began the streamlining and harmonization of the licences. In practice, the licensee must advise CNSC staff of any significant proposed modifications well in advance of the implementation. CNSC staff review these and encourage the licensee to take

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				of modifications implemented or planned that didn't comply with those requirements and consequently needed a written approval of the CNSC?	such steps as necessary to ensure a level of safety at least equivalent to that implied in the original licensing basis. This consultative approach, with iterations as necessary, has circumvented the need to have the Commission formally consider changes that would be outside the licensing basis.
24	Spain	Article 7.1	49	In the new, streamline operating license format, if the licensee or CNSC proposes to change the version of a particular regulatory document or standard that is cited in the license, the change can be executed by CNSC staff as long as the new version is at least as "safe" as the existing one. Have you developed guidance on the criteria to be applied for deciding that a "new version" is as "safe" as the existing one? How do you manage to harmonize the criteria and maintain consistency among CNSC staff practices?	The staff practice of updating versions of regulatory documents or standards that are in the licence has, in fact, not been used since CNSC began the streamlining and harmonization of the licences. Therefore, it has not been necessary to develop general criteria that could be used in repeated applications.
25	United States of America	Article 7.1	Pages 46-47	Explain how the observations and lessons learned from the Bruce A and Point Lepreau refurbishment projects were incorporated in the plans to refurbish reactors at Darlington?	The Darlington Refurbishment Program has sought out, gathered, and incorporated a significant amount of industry knowledge and experience pertaining to the planning and execution of major nuclear refurbishment and other mega projects including Bruce A and Point Lepreau rehabilitation projects.

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					OPG obtained this knowledge by receiving Operating Experience (OPEX) and Lessons Learned and actively seeking out OPEX and Lessons Learned through benchmarking visits, project and peer reviews, industry working groups (i.e. COG, CII), and involvement in WANO activities at Bruce Power, NB Power, Pickering A NGS, Pickering B NGS and Wolsong, Korea. Additional OPEX and lessons learned have been incorporated from benchmarking of non-CANDU NPPs and non-nuclear mega projects and incorporation groups such as the Project Management Institute, INPO's Project Management web sites and others.
26	Germany	Article 7.2.1	Annex, page 242	The factors to be considered in the graded approach are as follows: • the reactor power • the source term • the amount and enrichment of fissile and fissionable material • spent fuel elements, high pressure systems, heating systems and the storage of flammables, which may affect the safety of the reactor • the type of fuel elements • the type and the mass of moderator, reflector and coolant	Formal categorization of reactor types is not done in Canada because Canada's regulatory framework of requirements and guidance is intended to be broadly applicable to the full continuum of reactor types from research reactors to large NPPs. Although CNSC has two regulatory documents for design, one for NPPs (RD-337 "Design of New Nuclear Power Plants) and one for Small Reactors (RD-367 "Design of Small Reactor Facilities"), the requirements are for the most part the same. RD-367 provides some additional flexibility in the use of the graded approach for a number of areas concerning these smaller facility types. Each specific reactor project has its own

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				• the amount of reactivity that can be introduced and its rate of introduction, reactivity control and inherent and additional features • the quality of the confinement structure or other means of confinement • the utilization of the reactor (experimental devices, tests, reactor physics experiments) • siting, which includes proximity to population groups The graded approach –as described in IAEA Safety Standard Series No. NS-R-4 "Safety of Research Reactors"– shall be a structured method to balance the stringency of the requirement with the actual hazard potential of the reactor. Can Canada explain why the first step of the graded approach as described in NS-R-4 is not fully adopted and the categorization of the reactor based on its hazard potential is missing? Also, the second step to analyse the structures, systems and components to determine their importance to safety seems not to	unique safety case dependant on a number of factors such as those quoted from IAEA-NS-R-4 in the query from Germany. In a safety case in Canada, the proponent proposes in their safety case how they will meet Canadian requirements and conduct their proposed licensed activities in accordance with these requirements. CNSC then reviews the proposal (including the supporting evidence) taking into account factors such as those contained in NS-R-4 (amongst others). Hazard potential is part of that regulatory discussion between the proponent and the CNSC. As discussed in Article 7 (Definition of Licensing Basis) that proponent's proposal becomes part of the licensing basis for that project. Regarding the second question, analysis of structures, systems and components to determine their importance to safety is expected to be performed as part of the proponent's safety classification activities in the ongoing design process. Safety Classification, by definition, is a risk-informed process. RD-337and RD-367 both contain requirements that address classification of SSCs.

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				be taken into account.	
27	Indonesia	Article 7.2.1	p. 41/341 or p. 29	The NR mentions that The CNSC's regulatory regime defines NPPs as Class IA nuclear facilities and the regulatory requirements for these facilities are found in the CNSC Class I Nuclear Facilities Regulations. These regulations require separate licenses for each of the five phases in the lifecycle of a Class IA nuclear facility: a license to prepare a site, a license to construct, a license to operate, a license to conduct decommissioning and a license to abandon. Could you explain why in the regulations do not require a license to conduct commissioning during the lifecycle of a Class IA nuclear facility ?.	Clause 6(g) of the Class I Nuclear Facilities Regulations addresses commissioning. "(g) the proposed commissioning program for the systems and equipment that will be used at the nuclear facility;'" This is information to be submitted in an application for a licence to operate. However, through INFO-0756 "Licensing Process for New Nuclear Power Plants in Canada", and RD/GD-369 "Licence Application Guide, Licence to Construct a Nuclear Power Plant", CNSC permits fuel-out commissioning under a licence to construct. Commissioning with fuel loaded is only permitted under a licence to operate.
28	Japan	Article 7.2.1	p33	Canadian report says the regulatory documents are to be revised or amended to incorporate the lessons learned from Fukushima accident. Are the revised or amended documents to be back fitted or retroacted to the existing reactors?	The CNSC Fukushima Task Force report and Integrated Action Plan identified several improvements to the existing licensing basis, including guidance documents, regulatory standards and to two regulations. The Integrated Action Plan included both updates to the regulatory framework and documents, and actions taken by industry to implement recommended safety upgrades to

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					existing operating facilities. An omnibus regulatory document project was initiated to develop all necessary priority amendments in the relevant documentation with a very aggressive three year timeline for implementation by end Calendar 2014. Additionally, the CNSC implementation plan/strategy of revised/new REGDOCs is being integrated into the licensing basis for existing plants. For instance, the updated REGDOCs for Environmental Protection, SAMG and accident management are being added into licence conditions as appropriate in the power reactor operating licences (PROLs) to make them mandatory post-Fukushima, even though licensees are implementing these in response to the CNSC Integrated Action Plan. As the new REGDOC-2.4.1,"Deterministic Safety Analysis", and REGDOC-2.4.2, "Probabilistic Safety Assessment for NPPs" are completed for example, plans for their inclusion in the Licence Condition Handbook to be compliant with current versions are being initiated.
29	Korea, Republic of	Article 7.2.1	32, table 3(239)	In Page 239, Table 3 shows CNSC documents for NPPs that were developed using IAEA standards. Most of the "Associated IAEA standards" in column 2 were	Many of the regulatory documents listed in table 3 are currently being updated or included in the regulatory framework plans for review and update. As the new documents are developed, they are informed by or reference

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				revised (for example, NS-R-1 -> SSR-2/1). 1) Do you have any plan to revise the "CNSC documents" in column 1 of Table 3 using the revised IAEA standards? 2) If you have any plan, how much time and manpower are expected for the revision work?	the more recent IAEA documents. As an example, a new draft CNSC design document for NPPs is in development and now includes references to SSR-2/1. The CNSC has established the development and maintenance of its regulatory framework as a key corporate priority, and assigns resources based on priorities established by the corporate governance committee as outlined in the CNSC Regulatory Framework Plan. The CNSC has adopted a practice of reviewing all elements of its regulatory framework every five years or sooner if substantive issues or OPEX identifies a need for improvement. These reviews include a review of the latest developments in international guidance and best regulatory practice. The development of the regulatory framework plan considers the overall work with a multi-year view of the development and needed resources. The continuing update of the regulatory documents includes consolidation of documents into the modernized and streamlined regulatory framework structure to manage the risk, priorities, amount of work and resource availability. Additionally, the CNSC has strengthened its relationship with the nationally accredited standards system for the national consensus

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					standards program that supports the CNSC regulatory framework and licensing. This standards program also requires regular reviews and updates to address modern standards and methodologies, and international standards. The industry stakeholders lead the work program on a needs, risk and prioritization basis, and have enhanced its funding and resource support to manage the standards program growth.
30	Norway	Article 7.2.1	31	CNSC regulatory framework documents are very well structured and it seems that a lot of effort has been made in making these documents. It is reported that revision of these documents in future will be made as it is needed. It is also obvious that revision of regulatory documents is a continuous process with the modified good practices approach in the future. So the question is how it will be done in the future (if done in the past, how it worked) and how much manual and economic resources will be required for this. Do the CNSC have adequate resources available to perform the above said job efficiently?	The CNSC has established the development and maintenance of its regulatory framework as a key corporate priority, and assigns resources based on priorities established by the corporate governance committee as outlined in the CNSC Regulatory Framework Plan. The CNSC has adopted a practice of reviewing all elements of its regulatory framework every five years or sooner if substantive issues or OPEX identifies a need for improvement. These reviews include a review of the latest developments in international guidance and best regulatory practice. The development of the regulatory framework plan considers the overall work with a multi-year view of the development and needed resources. The continuing update of the regulatory documents includes consolidation of documents into the modernized and

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					streamlined regulatory framework structure to manage the risk, priorities, amount of work and resource availability. The consolidation of documents should also streamline the future resources needed to address regular reviews and incorporation of continuing lessons learned. Additionally, the CNSC has strengthened its relationship with the nationally accredited standards system for the consensus standards program that supports the CNSC regulatory framework and licensing. This standards program also requires regular reviews and updates to address modern standards and methodologies, and international standards. The industry stakeholders lead the work program on a needs, risk and prioritization basis, and have enhanced its funding and resource support to manage the standards program growth.
31	Russian Federation	Article 7.2.1	Annex 7.2	Appendix 7.2 to the Report states that in Canada a special regulatory approach is established for small power reactors. In particular, in setting forth the requirements the factors such as reactor power, amount of accumulated radioactivity, features of the primary circuit design and fuel handling system, type of fuel rods, type of moderator, reflector and	As a point of clarity, it is necessary to reinforce that there is no special regulatory approach for small power reactors (or Small Reactors as described in Annex 7.2 (i) (c)) Documents such as RD-367 “Design for Small Reactors” and RD-308 “Deterministic Safety Analysis for Small Reactor Facilities” contain, for the most part, the same requirements as those in the regulatory documents for NPPs (RD-337 “Design of New Nuclear Power Plants” and RD-310 “Safety Analysis for

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				coolant are considered. Would you provide additional information on how and in what documents of the regulatory body each of the listed factors is considered in formulating safety requirements for small power reactors?	Nuclear Power Plants”). The difference lies in where additional use of the graded approach may be applied in the interpretation of requirements. When formulating design and safety requirements for small reactors, CNSC staff is broadly guided by P-299 “Regulatory Fundamentals” (http://nuclearsafety.gc.ca/pubs_catalogue/uploads/P-299FinalPublicationApril05_e.pdf) which contains policy direction from the Commission on: • Section 4.1: setting requirements and assuring compliance • Section 4.2: basing regulatory action on risk In keeping with principles outlined in both Section 4.1 and 4.2 of P-299, staff ensures key principles in requirements and guidance take into consideration national standards and international practices such as requirements and guidance published by the IAEA. Specific to Small Reactors, IAEA NS-R-4 “Safety of Research Reactors” among others was considered when developing design requirements contained in RD-367 and safety analysis requirements in RD-308.
32	Argentina	Article 7.2.2	Article 7; Section III.7.2 (i); page 28	It is said that the CNSC updated its regulatory framework plan for the	The CNSC has established the development and maintenance of its regulatory framework

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				years 2012 to 2018 to outline the regulations and regulatory documents it will be developing or amending (Article 7; Section III.7.2 (i); page 28). It is also said that the peer-review team for the follow-up IRRS mission in 2011 concurred that the CNSC has developed a plan for systematic review of published regulations and regulatory guidance. Could you provide details about the mentioned plan for systematic review of published regulations and regulatory guidance?	as a key corporate priority, and assigns resources based on priorities established by the corporate governance committee as outlined in the CNSC Regulatory Framework Plan. The CNSC has adopted a practice of reviewing all elements of its regulatory framework every five years or sooner if substantive issues or OPEX identifies a need for improvement. These reviews include a review of the latest developments in international guidance and best regulatory practice. The development of the regulatory framework plan considers the overall work with a multi-year view of the development and needed resources. The continuing update of the regulatory documents includes consolidation of documents into the modernized and streamlined regulatory framework structure to manage the risk, priorities, amount of work and resource availability. The consolidation of documents should also streamline the future resources needed to address regular reviews and incorporation of continuing lessons learned. Additionally, the CNSC has strengthened its relationship with the nationally accredited standards system for the consensus standards program that supports the CNSC regulatory framework and licensing. This standards program also requires regular reviews and

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					updates to address modern standards and methodologies, and international standards. The industry stakeholders lead the work program on a needs, risk and prioritization basis, and have enhanced its funding and resource support to manage the standards program growth.
33	France	Article 7.2.2	7.2(ii)(d), 45	Reactors have a licence to operate, which has a period of validity (generally 5 years); hence, the licence needs to be renewed, notably by taking into account new regulatory documents or standards. An operating licence can also be amended by the Commission during its period of validity. In case of publication of a new regulatory document, what criteria are used by the Commission to choose between a licence amendment (i.e. immediate application) or a licence renewal (i.e. delayed application)?	Clause 8.(2) of the General Nuclear Safety and Control Regulations states that the Commission may, on its own motion (and among other things), amend a licence under any of the following conditions. (a) the licensee is not qualified to carry on the licensed activity; (b) the licensed activity poses an unreasonable risk to the environment, the health and safety of persons or the maintenance of national security; (c) the licensee has failed to comply with the <Nuclear safety and Control> Act, the regulations made under the Act or the licence; (d) the licensee has been convicted of an offence under the Act; (e) a record referred to in the licence has been modified in a manner not permitted by the licence; (f) the licensee no longer carries on the licensed activity; (g) the licensee has not paid the licence fee

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					prescribed by the Cost Recovery Fees Regulations; or (h) failure to do so could pose an unreasonable risk to the environment, the health and safety of persons or national security It is the last criterion that the Commission would use to decide if a licence amendment is necessary to add a new regulatory document. Typically, the licensee's provisions, in conjunction with the regulatory documents and standards already cited in the licence, are sufficient to address new developments or concerns. During the reporting period, the Commission used licence renewals, rather than licence amendments, to introduce new regulatory documents and standards in the licences to operate NPPs.
34	Germany	Article 7.2.2	page 38	The CNSC is executing a comprehensive plan for the preparation of licensing process documentation, regulatory documents and guides and application guides and forms. This plan includes the integration of knowledge gained from international licensing experience through organizations such as the IAEA, the Nuclear Energy Agency (NEA), the Multinational Design Evaluation Programme (MDEP)	Where relevant to a CNSC licensing matter, CNSC staff will consider other regulators' findings and conclusions and will discuss with them to understand the bases of their conclusions. It is well-recognized that certification in another country is informed by that country's laws and regulatory requirements and that any design must meet Canadian requirements, which are mature and well-developed. CNSC uses the approach of "trust but verify" when making use of specific findings from foreign regulators. For example, CNSC may

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				and other nuclear regulators. Typically, the Canadian regulatory body has to deal with licence application for CANDU reactors. Can Canada elaborate in more detail, how it can benefit from licensing procedures and assessments of foreign nuclear regulatory bodies?	refer to foreign regulatory reviews of specific aspects of designs and utilize a different approach or focus in its assessment. In the end, CNSC would compare its results with the foreign review to see if the conclusions were similar and, if not, determine the reasons for any differences.
35	India	Article 7.2.2	7.2 (ii)(a) & (b), Fig-7.2, Page-36, 41	CNSC Licence to prepare a site requires public hearing/information meetings to be held by the applicant. Whether such public hearing or information meeting is conducted for each phases of licensing as well as life extension of NPPs?	Paragraph 3(j) of the Class I Nuclear Facilities Regulations has a general requirement for licence applications for all life-cycle phases of Class I facilities (which includes NPPs) to include "…the proposed program to inform persons living in the vicinity of the site of the general nature and characteristics of the anticipated effects on the environment and the health and safety of persons that may result from the activity to be licensed." CNSC regulatory documents RD-346 Site Evaluation for New Nuclear Power Plants and RD/GD-369 Licence Application Guide: Licence to Construct a Nuclear Power Plant describe CNSC requirements and expectations for public information and consultation. As well, CNSC regulatory document RD/GD-99.3, Public Information and Disclosure was published in March 2012. It elaborates on the requirements and addresses the characteristics of the applicant/licensee's public disclosure protocols. It will be cited in all licences for

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					Class I facilities (as well as uranium mines and mills facilities and some Class II facilities). It is already cited in the licences to operate NPPs that were renewed since 2012, and is also cited in the licence to prepare a site for the proposed new-build at the existing Darlington site. Although the above CNSC regulatory documents do not explicitly require the applicant to hold public information sessions, they do have robust requirements for public engagement and information disclosure which would be addressed by various, diverse measures that typically include public information sessions. Currently, the expectation of both the regulator and the public in Canada is for meaningful information exchange and consultation when licensing decisions are made. Life extension of an NPP is part of a licensing phase and therefore any requirements for public information programs of the applicant are addressed through the process to renew the licence to operate the NPP.
36	India	Article 7.2.2	7.2 (ii), Page-36	What are the requlatory documents in which requirements for license to abandon are specified ? In case, it is not documented, can CNSC brief on different aspects to be checked before issuing licence to abandon.	Within the context of the *Nuclear Safety and Control Act* (NSCA) and its regulations, “abandonment” of a nuclear facility means that it is released from CNSC regulatory control and licensing. An applicant for a licence to abandon must submit the information required by sections 3 and 4 of the *General Nuclear Safety and Control Regulations*. Additionally,

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					if the application is in respect of a 'nuclear facility' that is defined by the legislation, sections 3 and 8 of the *Class I Nuclear Facilities Regulations* apply. If the application is in respect of a uranium mine or mill, sections 3 and 8 of the *Uranium Mines and Mills Regulations* also apply. A licence can be issued in only two situations. The first is when any residual nuclear substances that remain on site are below conditional or unconditional clearance levels established by the *NSCA* and defined through the *Nuclear Substances and Radiation Devices Regulations* (NSRDR). The other is when alternative arrangements are in place with other levels of government to ensure that the requirements of the *NSCA* and its regulations are being met (administrative controls). An application for a licence to abandon is normally applied for after the decommissioning project has been completed, the end state criteria met, and when final monitoring results confirm that it is acceptable to release the facility from CNSC regulatory control. The applicant will submit along with the information requirements identified above, decommissioning project summary reports, final monitoring data and follow-up program monitoring data that is often required by the environmental assessment process. No nuclear power plants have yet submitted an

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					application for a licence to abandon. However, Dalhousie University's SLOWPOKE-2 Reactor (DUSR) facility received a Licence to Abandon in August 2011, while in February 2014, a licence to abandon the Bruce Heavy Water Plant was issued by the Commission. Additional information on these facilities can be found in Canada's National Report for the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, Fourth Report
37	Russian Federation	Article 7.2.3	p.55	The Article says that the CNSC system uses 15 safe operation indicators (Appendix 7.2 (iii) (b)), which are regulated by the departmental document S-99 "Reporting Requirements for Operating Nuclear Power Plants". What are the selection criteria of these indicators? Was the international experience used in developing this indicator system (IAEA, WANO)?	These indicators reflect a combination of Canadian specific indicators and WANO indicators based on those that were in use in 2003. The CNSC is currently consulting with industry to update the set of safety performance indicators as part of REGDOC 3.1.1 which is expected to supersede S-99 "Reporting Requirements for Operating Nuclear Power Plants".
38	Russian Federation	Article 7.2.3	Appendix 7.2 (iii) (b), p. 245	Two indicators that assess water chemistry - "chemistry index" and "chemistry compliance index" – are in use. Please, explain what is the difference between the two indicators (their calculation	Chemistry Index measures performance of maintaining operational parameters against specification set by the licensee for equipment operability. Chemistry compliance index measures performance in complying with regulatory requirements set by the CNSC for safety.

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				methodology)?	
39	India	Article 7.2.4	7.2 (iv), Page-58	CNSC introduced new enforcement tool i.e. administrative monetary penalties (AMPs) to impose monetary penalties for the violation of a regulatory requirement e.g. environmental protection etc. What are the quantified guidelines for these violations to impose monetary penalty? Can licensee challenge CNSC decision in the court before paying the penalty?	The quantified guidelines for imposing these penalties are detailed in CNSC Regulatory document entitled Administrative Monetary Penalties Regulations. Within these regulations are found tables that define the specific regulation under applicable legislation, license conditions, or complimentary regulations for which an AMP could be applied. In practice this encompasses almost the entire regulatory framework of each of our licensees. Each individual requirement of the regulatory framework is assigned a risk category and these risk categories are then cross referenced to a table which defines the maximum and minimum penalty for that violation. Within the defined penalty range, the specific amount is determined by applying criteria to mitigate or aggravate the penalty amount starting at a base value. It should be noted, however, that AMPS are not applied indiscriminately; they are merely one more enforcement tool in a tool-kit of graduated enforcement measures that range from promotional activities to prosecution. As with all enforcement measures at the CNSC, they can be challenged, both formally or informally, depending on the circumstances. With respect to AMPs specifically, there is a formal procedure that licensees can use to

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					challenge/appeal the penalty. This process is published and simplified for use of smaller licensees and larger ones alike. In the absence of a win by licensees at the CNSC Commission level of resolution, there is always the opportunity to challenge the penalty in court. Since the AMPs program was brought into force last year there have only been three AMPs issued (to non-NPP licensees) and no challenges to date.
40	Indonesia	Article 7.2.4	p. 68/341 or p. 56	The NR mentions that during the reporting period, graded enforcement tools are available to the CNSC included the following: written notices, increased regulatory scrutiny, requests from the Commission for information, orders, licensing actions and prosecution. What are the process and requirements for suspension or revocation of a license? Is there any experience in doing these enforcements?	The graduated enforcement scheme at the CNSC is outlined in the CNSC Management System Manual process entitled “Enforce Compliance” which describes the process for acting in cases where compliance is unsatisfactory. The CNSC uses a graduated approach to enforcement, based on risk significance. The Commission may order licensees to appear before it, and may impose restrictions or revoke licenses. Essentially, all issues of serious non-compliance are brought to the attention of the Commission through formal written submissions and oral presentations by staff during a public hearing. Staff may make a recommendation for a license revocation or not. The licensee will also have the opportunity to be heard at these hearings and present their point of view. The Commission may, at their discretion, revoke a license in

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					support of – or contrary to - staff recommendations. A licence revocation should not be confused with a compliance order issued to curtail or restrict certain licence activities until the subject of the order is addressed. An order is issued by an inspector or designated officer and can be appealed to the Commission. A Commission decision can only be challenged in Federal Court. In Canada a license revocation is very rare, orders are not uncommon, but neither are they routine.
ARTICLE 8: REGULATORY BODY					
41	Argentina	Article 8.1	Article 8; Section 8.1 (c); page 74	The National Report mentions that "The Inspector Training and Qualification Programme entail the development and implementation of an effective, standardized and systematic approach for training and qualifying all CNSC inspectors. The program is composed of a combination of core training, service-line specific training and on-the-job training" (Article 8; Section 8.1 (c); page 74). Does the programme include any mechanism to categorize inspectors and in that case, how are the categories accredited?	The first inspector categorization occurs after an individual has successfully completed an extensive training program, on-the-job-training and review of skills through written examination for certain modules. Successful candidates are provided credentials in accordance with the Nuclear Safety and Control Act and regulations. Further categorizations tend to be organizational, job classifications with some Inspectors being appointed to the role of Site Supervisors who in addition to being certified inspectors act as mentors for site inspections teams.

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42	Argentina	Article 8.1	Article 8; Section 8.1 (d); page 77	Could you mention, if it is possible, some indicators to measure Management System effectiveness of the CNSC? (Article 8; Section 8.1 (d); page 77)	The CNSC primarily uses qualitative and anecdotal performance indicators for measuring and improving the effectiveness of the CNSC Management System. Examples of such measures include staff awareness of roles and responsibilities, regulatory activities such as inspections and desktop reviews completed (vs. planned); findings arising from formal and informal reviews, audits and assessments; insights and feedback gained through surveys of staff and of stakeholders; feedback gained during our interactions with licensees and numerous outreach activities; feedback gained from award applications; etc. All Government of Canada agencies, including the CNSC are assessed periodically against management excellence-related criteria set out by the government's Management Accountability Framework (MAF). The last MAF assessment was conducted in 2009 and the next assessment is scheduled for 2014. The most recent comprehensive assessments of the effectiveness of the CNSC management system were the 2009 IRRS mission and the 2011 follow-up mission. The IAEA assessed the extent of alignment of the CNSC's management system against the IAEA safety standard GS-R-3 Management System for Facilities and Activities and concluded that "overall, Canada has a mature and well-established nuclear regulatory framework and that the nuclear regulator does an effective job

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					in protecting the health, safety and security of Canadians and the environment." and that "The CNSC has done extensive and commendable work over the last years to develop the Management System."
43	Argentina	Article 8.1	Article 8; Section 8.1 (f); pages 81 to	Please, could you give more information about the concrete operational working mechanism of the interchange and consultation with stakeholders, the participant funding program, choice of eligible interveners, and how you take into consideration the exchanged information and how binding it is? (Article 8; Section 8.1 (f); pages 81 to 83)	The CNSC has a very strong consultation approach, where it will involve interested members of the public, non-governmental organizations and other departments in the development of policies, regulatory documents and standards, proposed regulations and regulatory amendments and possible legislative changes. In terms of process, the commission usually issues a proposed document and invites comments with a period of 60 to 120 days. Once the comments are received, these are posted with a possibility of commenting on the comments. A comments disposition table is usually prepared to indicate how the comments were addressed, and a revised document published (as final if comments not substantive) or for further comments (if comments were substantive). Comments received are not binding, but are duly considered and the CNSC responses communicated. In the same vein but differently, there is a statutory requirement in the NSCA to provide an opportunity to be heard in the context of licensing hearings. This process is more formal than the consultations referred to above,

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					as the Act and Rules of Procedure provide a more rigorous process for applicants to provide their evidence and for members of the public, NGOs, etc. (the intervenors) to provide their comments in writing and possibly orally. The Rules of procedure provides the discretion to the Commission to permit or refuse interventions. The Rules (Rule 19) specify that a person should have an interest in the matter or has expertise that may be useful to the Commission in coming to a decision. In practice, most of the interventions are permitted. All intervenors must file a written submission usually at least 30 days prior to the hearing, and if they so request are provided 10 minutes to make an oral presentation (followed by questions from the Commission members). The interventions are part of the record (the CNSC is a court of record) and are considered by the Commission members in their deliberations. The Commission’s final decisions will refer to key matters raised by intervenors. With respect to the Participant Funding Program, this is available to all potential intervenors, but must usually be linked to a commitment to provide value-adding submissions to the hearing process. For example, funding received could be used to assist affected intervenors in remote communities to participate in the process or to hire experts or specialists to assist in better

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					understanding the potential impact of a new or expanded uranium mine near their community. Available amounts are limited.
44	China	Article 8.1	P 64	This section and Annex 18 (i) both mentioned the new neutron-overpower methodology (ROPT) and the installation in OPG NPP. Does that mean that CNSC has approved the new ROPT method? Could you please introduce the position and attitude about the new method and the installation situation in OPG NPP?	The CNSC has not approved the new neutron overpower (NOP) analysis methodology for regulatory applications. However, the CNSC has approved a 1% increase in the installed NOP trip set-points at Darlington NGS due to the increase in critical channel power (CCP) gained from the implementation of the modified 37 element fuel bundle design.
45	China	Article 8.1	15	This section and Annex 18 (i) both mentioned the installation of a third Class III electrical power standby diesel generator (SDG) in Point Lepreau. Is the additional SDG seismic-qualified or not? Does that mean that the other two SDGs can be maintained preventatively during the power operation? Could you please give some detailed introduction about the third Class III SDG?	The third Class III standby generator is an installed spare and is two half-size units that work together to produce the full rated power capability. The installed spare is only used and tied to the electrical distribution bus when one of the other two standby generators is unavailable; thereby maintaining the design basis of two available standby units.
46	Ireland	Article 8.1	Article 8.1. (f), p 83	Openness and transparency - how does CNSC measure the performance of its communication	A variety of methods are used to assess the performance of our communication and educational tools. These include:

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				and education tools?	• Web and social media metrics • Ongoing feedback from the stakeholders • Feedback from staff who use our products to conduct outreach activities • Performance against service standards (for responding to public inquiries) • Benchmarking against other organization (e.g., number of views for our videos) • Environmental analysis
47	Korea, Republic of	Article 8.1	67	It is understood that decision making for major issues are being carried out through Commission Tribunal. 1) Please describe specific decision making process, and if a number of measure are being used for the decision making of an issue, please describe the measures being used and how the measure are being interacted with other measures. 2) If a number of measures are being used for the decision making of an issue, please describe the differences in the binding force of the measures. 3) Also, please describe that if there is any contradiction between	Decision-making by the Commission tribunal component is mostly on licensing matters pertaining to larger licensees (nuclear power plants, fuel manufacturing, uranium mines and mills). (1) The licensing process typically entails the following: • receipt of an application by an applicant/licensee; • many exchanges between CNSC staff and the applicant to address concerns and questions from CNSC staff; • the publication of a Notice of Hearing; the filing by the applicant of a formal Commission Member Document (CMD) synthesizing their arguments in support of a licence; • the filing by CNSC staff of a CMD

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				certain measures during a decision making process, how they are being resolved? 4) Please explain how the follow-up activities or processes of decisions that has been made are being carried out and who will be responsible for the follow-up activities or process, the Commission or a certain branch or a staff.	incorporating their recommendations (and the basis for these) to the Commission members whether to issue a licence, and licence conditions thereof; • the conduct of Part 1 of a public hearing where the applicant and CNSC staff will present their respective CMDs in a public forum; • 30 days later, the filing by public intervenors of their CMDs; • 30 days later the conduct of Part 2 of the public hearing where public intervenors present their submissions and Commission members ask questions to the applicant, CNSC staff and public intervenors on the evidence presented for consideration; • 60-90 days later, the Commission issues its final decision. There is no interaction with other measures, unless one considers that all other licensing and compliance measures are made by CNSC staff (designated officers, etc.) following less complex processes (these represent the large majority of decisions). (2) Commission decisions are binding as they set out the terms and conditions of licences. Compliance programs are in place to ensure compliance and conformity with the statutory, regulatory and licensing requirements. Failure

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					to comply may give rise to a number of compliance or enforcement options, from action notices to binding orders to fines to revocation of licences, etc. (3) The role of the Commission tribunal is to make a decision based on all the evidence presented. If there is conflicting or contradictory submissions (for example, the applicant and CNSC staff disagree on some technical issues or licence conditions), the Commission tribunal will evaluate the information before it, and will render a binding decision. An applicant would then have to abide with the decision, or ask for a judicial review of the decision by the Federal Court of Canada (applicants have never applied for judicial review of a decision). (4) The follow-ups are mostly under the responsibility of CNSC staff. This being said, CNSC staff must report annually to the Commission, in a public proceeding, to report on the performance of each major nuclear facility. Comprehensive inspection programs are in place to ensure conformity with statutory, regulatory and licensing requirements, and inspectors are located at major facilities, supported by a large team of technical specialists.
48	Korea, Republic of	Article 8.1	63	It is described that the members of the CNSC Commission ¡°are chosen on the basis of their	Members of the Commission are subject to stringent conflict of interests provisions (could be provided upon request). In the example

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				credentials and are independent of all political, governmental, special interest group or industry influences¡±. However, the Canada¡¯s response to the Question No. 7 (by Ireland) in the ¡°Responses to Questions Raised from Peer Review of Canada's Fifth National Report for the Convention on Nuclear Safety stated that "For example, the current Members of the Commission include a business person who is also a former provincial energy minister" (page 4). Does the CNSC consider that such former position (energy minister) does not create any conflict-of-interest with the current position of that CNSC Member of the Commission?	provided, the former Energy Minister was a minister in a provincial/local government, not the federal Canadian government. As nuclear energy is under the sole jurisdiction of the federal government, there was no conflict. In any event, this part-time member had been retired for several years prior to his appointment.
49	Korea, Republic of	Article 8.1	78	It is understood that the CNSC has made a progress with RIDM. 1. When and by whom is it decided to initiate the RIDM process? Was screening criteria established? 2. It is mentioned in Appendix H that socio-economic implications	1. The RIDM process is used to ensure a balanced perspective in cases where there are multiple factors to be taken into consideration. The decision to initiate the process is made by the decision maker, who oversees the process to ensure that the team has all required resources available to them and a complete assessment is carried out. 2. The licensing basis set by the Commission

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				are considered. How can socio-economic implications be considered without a compromise to safety? Did the CNSC seek consent from the public that a small increase of risk would be allowed as far as safety goal is achieved?	includes safety goals which licensees are required to meet. In applying RIDM, multiple alternatives are often evaluated. Each alternative must meet the safety goals included in the licensing basis, but beyond that other factors (such as socio-economic) may contribute to the selection of which alternative is chosen.
50	Romania	Article 8.1	Art. 8.1 (c)	In chapter 8.1 (c) it is mentioned that he CNSC continued to contribute to CANTEACH and University Network of Excellence in Nuclear Engineering programs. How does the CNSC make use of CANTECH in the training programme of its staff?	CANTEACH is a knowledge repository that provides high quality technical documentation relating to the CANDU nuclear energy system. This information is public and is intended for use in various aspects of education, training, design and operation. The CNSC contributes material to the 8 courses that make up the CANDU program (CANDU Fundamentals, Instrumentation and Control, Mechanical, Electrical, Heat and Thermodynamics, Chemistry, Fluid Mechanics, Reactor Boilers and Auxiliary). As a result, while the use of CANTEACH itself is not a formal part of the CNSC's corporate training programme, CNSC staff may receive training using much of the same or similar material on an as-needed basis. All CNSC staff members have Individual Learning Plans (ILPs) which, in following their supervisors' direction, could include direct use of CANTEACH. Such use of CANTEACH would therefore be part of an individual, rather than a CNSC corporate,

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					training program.
51	Russian Federation	Article 8.1	Appendix 8, para 4	Would you provide clarification information on the content of criteria for NPP resistance to external events, which were used in safety assessment of the Canadian NPPs after the accident at Fukushima-Daiichi NPP mentioned in Appendix 8?	As per CNSC Integrated Action Plan, licensees were requested to complete the review of the basis for external events against modern state-of-the-art practices for evaluating external events magnitudes and relevant design capacity for these events, including but not limited to: earthquake, floods, tornadoes and fire. The closure criteria identified for this action is to complete a re-evaluation, using modern calculations and state-of-the-art methods, of the site-specific magnitudes of each external event to which the plant may be susceptible. Licensees are to submit an evaluation for all hazards, including magnitudes of external hazards that were not screened out (as per Regulatory Document S-294 hazard evaluation and screening) The list of external hazards that the plant can be subject to needs to be identified in accordance with IAEA Safety Standard SSG-3, “Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants”, 2010. The hazards screening analysis can proceed in accordance with the guidance in SSG-3; however, in order to fully address Fukushima lessons learned, the following external hazards need to be evaluated at an appropriate

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					magnitude: • seismic hazard • external flooding hazard • high wind and tornadoes, which includes missile impact analysis Site specific consequential events from the analyzed external hazards need to be assessed (for example, consequential fire or flood from a seismic event).
52	Spain	Article 8.1	69	In relation to the inspection activities, could you specify the average number of inspections per plant and year, as well as the estimate resources (hours per person), in the case of a good performer plant, including those of the on-site CNSC office.	The CNSC’s compliance baseline presently indicates 70 compliance entries covering 14 Safety and Control Areas that are conducted over a five year rotating period. Supporting those specific activities there are several additional system inspections. Each inspection involves varying scopes and effort. We do not formally keep or track average inspection numbers per site.
53	Spain	Article 8.1	69	How many on-site inspectors have the CSNC per site? Is it established a time limit for an inspector to be assigned to a specific site?	There are four sites in Canada where inspectors are co-located with the utility. The on-site inspection staffing levels in 2013 were: Single Unit Station Gentilly-2: 4 Single Unit station Point Lepreau: 4; Multi Unit (8 reactors) Bruce Site: 8; Multi Unit (8 reactors) Pickering: 11; Multi Unit (4 Reactors) Darlington: 9; Movement of inspectors is not mandatory;

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					however, inspectors re-locating from site to site are not uncommon. Frequently, topic specific inspections are conducted using an inspector from a different site to inspect that specific topic. This practice is common and is encouraged by the CNSC.
54	Spain	Article 8.1	76	A new revision of the Management System Manual was planned for 2013. Has this new revision been approved?	The CNSC Management System Manual has been revised to update and improve the content. The document is currently being reviewed by management – target release date is now April 2014.
55	Switzerland	Article 8.1	p.60	Canada's nuclear regulatory body, the CNSC, strives for regulatory excellence. Are there any tools implemented for benchmarking? What criteria are used to identify excellence?	To better understand requirements and expectations associated with regulatory excellence, the CNSC aligns itself with all applicable IAEA safety standards. In addition, peer reviews such as IRRS missions, along with national and international conferences and workshops and one-on-one visits, provide opportunities for CNSC staff to gain more insights into how best to improve regulatory effectiveness. The CNSC also works collaboratively with other Government of Canada agencies to further refine its regulatory and supporting policies and processes and adapts such to the Canadian landscape. Depending on the availability of data, the CNSC conducts comparative analysis which may in turn, lead to more formal discussions and a review of underlying processes and/or

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					systems. All planned improvements simultaneously consider the people, process and infrastructure components to assure that needed change is understood, realized and sustained.
56	United Kingdom	Article 8.1	8.1b, d	How is the assurance of the correct application of regulatory processes, as exemplified in the Management System, Process Documentation and Work Instructions determined by the CNSC and how many staff are involved in this assurance programme?	The responsibility for assuring the correct application of regulatory processes is shared by a number of divisions across the CNSC. Training plays an important role in assuring staff knows, understands, and are capable of fulfilling their expected roles and responsibilities. Training requirements are identified and addressed throughout the process design, validation and implementation phases. To assure that our collective regulatory processes remain as effective and efficient as intended, the CNSC conducts informal reviews as well as formal audits and evaluations (internal and/or external by 3rd-party). At the local level, process performance is compared against expectations. Observed sub-standard performance and/or a desire to verify the correct application of process steps may lead to a more formal review or assessment. Self-assessments at the process level can be requested by the manager and are conducted with the assistance of staff within the Internal Quality Management Division.

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					Formal audits and evaluations are typically scheduled on a priority basis at the direction of the President. Findings are reported directly to the President along with associated management responses. Commitments to address shortcomings are detailed in approved management action plans with progress monitored through to completion and close-out.
57	United Kingdom	Article 8.1	8.1c	Once an Inspector has completed the Inspector Training and Qualification Program and received an Inspector's Card, what requirements are in place to maintain and demonstrate continued competence?	Once an individual has been designated as an inspector, they are issued an inspector certificate that typically is valid for 5 years. Annually, Directors meet with their inspectors to identify performance and training needs to maintain job proficiency. Required training is tracked via individual learning plans. Some service line specific training requires refresher training to maintain competencies. The inspector's effectiveness is also ensured through oversight by his Director and supervisor/coordinator.
58	United Kingdom	Article 8.1	8.1g	How do CNSC Inspectors engage and co-operate with Licensees' internal regulatory functions in order to support improvements in safety performance? Please provide information on any protocols that may be in place to support such interactions.	There are CNSC resident inspectors at Canadian NPPs, unlike most member countries. This proximity and availability to operators allows CNSC to have a very thorough understanding of the management of operations, compliance and licensing at the plant. The open door policies at plants allow staff to communicate effectively at the stations, on various topics including the interpretations of expectations and requirements. This also

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					allows for rapid communications and quick response to issues and events. A clear correspondence protocol between CNSC internal divisions and the licensee ensures that the correct parties have the right information in a timely manner and any expectations imposed on licensees fairly represent CNSC views under an official point of contact, while allowing and encouraging informal discussions between CNSC and licensee technical staff. Safety Performance at the station remains the responsibility of the licensee.
59	United Kingdom	Article 8.1	8.1d	Please provide information on the extent to which CNSC makes use of Technical Support Contractors to support its regulatory operations and provide a characterisation of the type of work undertaken and how this work is used by inspectors in making regulatory decisions.	The CNSC has its own "Technical Support Organization" referred to as the Technical Support Branch (TSB). This Branch is responsible to provide technical advice and support to the Commission, Designated Officers, Regulatory Program (licensing) Divisions and Inspectors. Inspectors regularly call upon this expertise in the conduct and assessment of inspections. The TSB has over 250 personnel, most with advanced degrees, who are organized into various areas of expertise: Engineering and Design, Safety Analysis, Management Systems, Environmental and Radiation Protection and Security and Safeguards. The TSB regularly contracts to external subject matter experts to supplement their expertise, as

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					required, in order to manage workload or to provide specialized expertise. Furthermore, the CNSC does acquire independent scientific expertise to support regulatory decision-making through its Regulatory Research Programs. Objectives of this research typically include verification and validation of licensee research, assistance in the identification of operational problems, to help develop capability and tools to support assessments, to develop and support safeguards approaches and technologies, and to aid in the development of safety standards. This research is overseen by specialists in the appropriate TSB division and their advice is provided to the Regulatory Operations Branch divisions responsible for regulatory oversight of power reactors. They will, in turn, weigh the information and request licensing changes and/or inspection activities, as necessary, based on a risk-informed approach to addressing the issue at the stations.
60	United Kingdom	Article 8.1	11.2b	Article 11 outlines the demographic challenges affecting the electricity sector in Canada, is CNSC affected by the same demographic challenges and what steps are being taken to address them?	CNSC faces workforce demographic challenges similar to our counterparts in industry: • Retirement continues to be a serious and impending issue. • Although 25% of our current workforce will be eligible to retire by March 31, 2017, 13% of our current workforce is expected

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					to retire within the next 4 years, based on a mean take up rate of 23% per-year. • Approximately 70% of CNSC employees are over 40 years old. Changes in workforce demographics, coupled with a shift in human capital competency requirements (e.g., due to a shift in focus to refurbishments and decommissioning) have led CNSC to develop and implement initiatives in four related areas, comparable to those of NPP licensees: • Detailed workforce capabilities analyses that clearly articulate core competencies required, optimal organizational design and identify critical roles. • Hiring programs that focus on renewal at the entry and mid-career levels. • An integrated training program for inspectors. • Knowledge retention programs in the form of robust succession plans, a successful Alumni program and transition funding earmarked to enable knowledge transfer within the workforce.
61	United States of America	Article 8.1	Section 8.1(c)	The CNSC plans to reduce staffing levels, as reactors enter the decommissioning phase, while, "…identifying critical positions…" for succession	Historically, CNSC reviewed and updated its critical positions matrix bi-annually. The matrix uses risk factor ratings to validate criticality of positions and identifies potential vulnerabilities. Mitigation strategies (e.g.,

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				planning. Please provide additional information on what positions and skills are critical with regard to expected future activities.	succession plans, high potential identification and development) are then developed and/or monitored based on the matrix output. Due to a rapidly evolving landscape, CNSC has formed a Strategic Workforce Planning working group to specifically explore possible future scenarios and identify key roles, core competencies and determine, with the help of external expertise, the optimal organization design required to meet the future demands of our industry. Furthermore, in the context of this work, the CNSC will examine its talent management strategy to ensure that it has the necessary supports and professional development in place to support those in these key positions.
62	China	Article 8.2	P78	RIDM is very important for CANDU plant to enhance the safety level. Could you please provide a full list of all these 73 safety issues?	The list of CANDU Safety Issues is as follows: #1 - GL 1: Classification of components #2 - GL 2: Environmental qualification of equipment and structures #3 - GL 3: Ageing of equipment and structures #4 - GL 4: Inadequacy of reliability data #5 - GL 5: Need for performance of plant-specific probabilistic safety assessments (PSA) #6 - RC 1: Inadvertent dilution or precipitation of poison under low power and shutdown conditions

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					#7 - RC 2: Fuel cladding corrosion and fretting #8 - CI 1: Fuel channel integrity and effect on core internals #9 - CI 2: Deterioration of core internals #10 - CI 3: SG tube integrity #11 - CI 4: Loads not specified in the original design #12 - CI 5: Steam and feedwater piping degradation #13 - PC 1: Overpressure protection of the primary circuit and connected systems #14 - PC 2: Safety valve and relief valve reliability #15 - PC 3: Water hammer in feedwater and steam lines #16 - SS 1: ECCS sump screen adequacy #17 - SS 2: Potential problems in ECCS switchover to recirculation #18 - SS 3: Severe core damage accident management measures #19 - SS 4: Leakage from systems penetrating containment or confinement during an accident #20 - SS 5: Hydrogen control measures during accidents #21 - SS 6: Reliability of motor-operated and

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					check valves #22 - SS 7: Assurance of ultimate heat sink #23 - SS 8: Availability of the moderator as a heat sink #24 - ES 1: Reliability of off-site power supply #25 - ES 2: Diesel generator reliability #26 - ES 3: Reliability of emergency DC supplies #27 - ES 4: Control room habitability #28 - ES 5: Reliability of instrument air systems #29 - ES 6: Solenoid valve reliability #30 - IC 1: Inadequate electrical isolation of safety from non-safety-related equipment #31 - IC 2: I&C component reliability #32 - IC 3: Lack of on-line testability of protection systems #33 - IC 4: Reliability and safety basis for digital I&C conversions #34 - IC 5: Reliable ventilation of control room cabinets #35 - IC 6: Need for a safety parameter display system #36 - IC 7: Availability and adequacy of

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					accident monitoring instrumentation #37 - IC 8: Water chemistry control and monitoring equipment (primary and secondary) #38 - IC 9: Establishment and surveillance of setpoints in instrumentation #39 - CS 1: Containment integrity #40 - IH 1: Need for systematic fire hazards assessment #41 - IH 2: Adequacy of fire prevention and fire barriers #42 - IH 3: Adequacy of fire detection and extinguishing #43 - IH 4: Adequacy of the mitigation of the secondary effects of fire and fire protection systems on plant safety #44 - IH 5: Need for systematic internal flooding assessment including backflow through floor drains #45 - IH 6: Need for systematic assessment of high energy line break effects #46 - IH 7: Need for assessment of dropping heavy loads #47 - IH 8: Need for assessment of turbine missile hazard #48 - EH 1: Need for systematic assessment of seismic effects

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					#49 - EH 2: Need for assessment of seismic interaction of structures or equipment on safety functions #50 - EH 3: Need for assessment of plant-specific natural external conditions #51 - EH 4: Need for assessment of plant-specific man induced external events #52 - AA 1: Adequacy of scope and methodology of design basis accident analysis #53 - AA 2: Adequacy of plant data used in accident analyses #54 - AA 3: Computer code and plant model validation #55 - AA 4: Need for analysis of accidents under low power and shutdown conditions #56 - AA 5: Need for severe accident analysis #57 - AA 6: Need for analysis of total loss of AC power #58 - AA 7: Analysis for pressure tube failure with consequential loss of moderator #59 - AA 8: Analysis for moderator temperature predictions #60 - AA 9: Analysis for void reactivity coefficient #61 - MA 5: Degraded and non-conforming conditions and operability determinations #62 - MA 13: Availability of R&D, technical

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					and analysis capabilities for each NPP #63 - OP 1: Operating experience feedback #64 - PSA 2: Equipment qualification #65 - PSA 3: Open design of the balance of plant - steam protection #66 - PSA 4: PHT relief #67 - PF 9: Fuel behaviour in high temperature transients #68 - PF 10: Fuel behaviour in power pulse transients #69 - PF 12: GAI 00G01 Channel voiding during a Large LOCA #70 - PF 15: GAI 95G01: Molten fuel/moderator interaction #71 - PF 18: Fuel bundle/element behaviour under post dryout conditions #72 - PF 19: Impact of ageing on safe plant operation #73 - PF 20: Analysis methodology for NOP / ROP trips
63	Japan	Article 8.2	8.2(a), p84	Canadian report describes separation of Commission members from the promotion side. What kind of provision does exist for assurance of separation of CNSC staff from the promotion side, such as, limitation of	The CNSC does not prevent employees from going to or coming back from the promotion side. While we have no formal provisions from an HR policy perspective, the CNSC Conflict of Interest and Post-employment Policy and Program, managed by our Office of Audit and

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				redeployment to/ from the promotion side?	Ethics, guide staff actions in this regard.
64	Korea, Republic of	Article 8.2	85	Through the initial and follow-up IRRS mission to Canada, the independence of the CNSC from NRCan was assessed on the several points and was confirmed to meet the requirements of IAEA GS-R-1 as described on the page 85 of the National Report. The points of assessment were mainly concerned with the relationship between the CNSC and NRCan. It is understood from the page 65 of the National Report that the Governor in Council designated NRCan as the administrative channel for the CNSC to report to Parliament and to seek funding support from the Government. Please explain why NRCan was chosen as the designation. Is it possible to change the designation? If the relationship is totally independent and it's possible to change the designation, why didn't Canada try to change the designated minister in order to dispel any issue, even minor, about the relationship?	The CNSC and NRCan are independent organizations. The CNSC reports to Parliament through the minister for administrative purposes, but acts fully independently in regulatory matters. This was the clear intent of the Canadian Parliament when the Nuclear Safety and Control Act (NSCA) was passed, and the CNSC established. According to the NSCA, the CNSC is established as an independent Tribunal, whereas NRCan is a government department. The NSCA and budgets for the CNSC are administratively managed through the minister, and its key regulatory activities are independent. The regulatory instruments, and plans and budgets of the Commission are developed and managed separately and independently to support its necessary operations. The CNSC's independent Commission Tribunal has the authority to independently approve facility and activity licences. It can also establish regulations and other regulatory instruments under the Act for final approval by the Governor in Council, independently from NRCan. Because the organizations effectively acts independently, it has not been seen as

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					necessary to change the relationship, or to change the designated minister.

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ARTICLE 9: RESPONSIBILITY OF THE LICENCE HOLDER					
65	India	Article 9	9(b), Page-88	It is mentioned that as post-Fukushima safety enhancements, on-site and off-site power supplies, pumps etc. are provided. Please clarify if these provisions are unit wise or are they shared among different units at a site?	The CNSC Integrated Action Plan calls for each site to acquire the necessary equipment for emergency back-up power and ancillary equipment required to maintain core cooling and spent fuel bay cooling in the event of prolonged station blackout. This equipment is maintained both on and off site in case of onsite equipment unavailability. For multi-unit NPPs, sharing of emergency equipment among the units is allowed. For example, an emergency generator could supply all units of a multi-unit. In the Province of Ontario, licensees have pooled some resources in a Regional Center which through mutual aid agreements between licensees may be used to supplement onsite assets should these required in an emergency. An Ontario based Regional Emergency Response Support Center (RERSC) is being pursued and will be available to all Canadian NPP licensees. The RERSC was one of the early industry-wide Fukushima OPEX recommendations. The primary goal of an RERSC is to house Emergency Mitigation Equipment that can be safely stored offsite and delivered immediately once site access is restored - or after the initial 3-day site self-sufficiency period

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66	Spain	Article 9	88	Are the licensees obliged to maintain a program to encourage the workers to identify and communicate any safety related deficiency and to protect the whistleblowers against retaliation?	All licensees are required to have programs under their Management Systems to identify and resolve any safety related deficiencies; these are generally integrated into the licensee corrective action program. This is covered by a condition in the licence to meet the requirements of CSA Standard N286-05 “Management System Requirements for Nuclear Power Plants”. Licensees also are required to provide protection to whistleblowers under the Nuclear Safety and Control Act and other Canadian legislation.
67	Spain	Article 9	92	Does CNSC have a program to manage communications on safety related deficiencies of the NPP reported by plant workers or the public?	As part of its process to manage public inquiries, the CNSC has established a procedure in order to manage whistleblower complaints. Such complaints are confidentially directed to the attention of upper management for further investigation. In matters of public disclosures of events at NPPs – The CNSC has established regulatory requirements for NPP operators to have robust public information and disclosure programs supported by disclosure protocols (see pages 91 and 92 of Canada’s 6th CNS report). In addition, the CNSC readily answers questions about events at NPPs from members of the public and media outlets CNSC staff also regularly reports on these events at public Commission meetings.

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ARTICLE 10: PRIORITY TO SAFETY					
68	Czech Republic	Article 10	Page 102	The report states that „The CNSC published a discussion paper entitled Safety Culture for Nuclear Licensees in August 2012. The discussion paper sets out the CNSC's overall strategy for safety culture in the Canadian nuclear industry, which comprising the following three components. Consultation on this discussion paper enabled the CNSC to engage with the industry, stakeholders and public on issues affecting safety culture. CNSC staff are currently analyzing and considering feedback on the discussion paper. Does the CNSC plan to develop the discussion paper further into a more formalized document or even its possible transposition into a regulatory guide/document in the future?	Yes, the CNSC's intention is to create a Safety Culture regulatory document that will consist of requirements and guidance for Canadian nuclear licensees. Stakeholder and public feedback acquired through Discussion Paper 12-07 in September 2013 will assist in the creation of the Safety Culture regulatory document. Public feedback can be found at: http://www.nuclearsafety.gc.ca/eng/acts-and-regulations/consultation/completed/dis-12-07.cfm
69	France	Article 10	§ 10.4.1.2 - p.118 to 120	How does Canada ensure that contractors and subcontractors of licensees maintain a positive safety culture and a clear understanding of the importance of safety first? Are there contractors involved in the assessments	When NPP licensees perform assessments, it is the CNSC's expectation that "all workers" participate including contractors. As well the CNSC's documentation for new builds sets out clear requirements on licensees with respect to safety culture, and explicitly states that safety culture is applicable to all

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				performed by licensees, such as the ones performed at Bruce Power based on the NSCMP methodology and process? Is there any requirement for contractors to conduct safety culture self-assessment as is the case for licensees?	personnel including contractors and sub-contractors.
70	France	Article 10	10 (b), 98	For safety culture self-assessment, the report states that the Nuclear Energy Institute guideline "has been adopted by most Canadian NPP licenses". What about those which have not?	The only Canadian NPP licensee that has not adopted the NEI guideline is Hydro-Quebec and given that utility has decided to permanently shutdown its Gentilly-2 nuclear power plant, they will not be implementing the guideline.
71	Korea, Republic of	Article 10	b, 96	It is stated in the paragraph 2.32 of IAEA Safety Guide GS-G-3.1 that the management system should establish a working environment in which staff can raise safety issues without fear of harassment, intimidation, retaliation or discrimination. Does the CNSC licensing requirements for management systems include this IAEA recommendation as a requirement? If so, how does the CNSC verify that such a working environment has been established and maintained? Does the CNSC have a process to deal with safety allegations from employees	Question 1: Do the CNSC licensing requirements for management systems include this IAEA recommendation as a requirement? Answer 1: The Canadian standard for management systems, CSA N286-12 “Management System Requirements for Nuclear facilities”, which was published in 2012 and is coming into effect this year as a license condition, has as a first principle that “Safety is the paramount consideration guiding decisions and actions”. This is supported by a requirement for Safety Culture which has as four criteria, two of which say “Management shall use the Management System to understand and promote a safety culture by: a) providing the means by which the business

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				working in nuclear industry?	supports workers in carrying out their tasks safely and b) monitoring to understand and improve the culture.” Question 2: If so, how does the CNSC verify that such a working environment has been established and maintained? Answer 2: The CNSC has expectations that licensees foster a healthy safety culture within their organization. In order to achieve this, licensees are expected to self assess. CNSC performs oversight of licensee’s self assessments. The self assessment does include this element. Question 3: Does the CNSC have a process to deal with safety allegations from employees working in nuclear industry? The Canadian Nuclear Safety and Control Act in section 48(g) prescribes that "Every person commits an offence who takes disciplinary action against a person who assists or gives information to the Commission, designated officer or inspector." There are various legal remedies available under Section 51(3) of the NSCA: “(3) Every person who commits an offence other than an offence in respect of which subsection (1) or (2) applies: (a) is guilty of an indictable offence and liable to a fine not exceeding $1,000,000 or to

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					imprisonment for a term not exceeding five years or to both; or (b) is guilty of an offence punishable on summary conviction and liable to a fine not exceeding $500,000 or to imprisonment for a term not exceeding eighteen months or to both." Alternately, the CNSC could make use of newly created Administrative Monetary Penalties, which in this instance could impose penalties ranging from 1000 to 40000 dollars, depending on the circumstances of the offence.
72	Pakistan	Article 10	Page-99, para-2	Canada may like to share the components of safety culture survey?	Yes, we can share the components and method. However, results are confidential.
73	Spain	Article 10	96	Does the Canadian NPP conduct periodic external assessment of safety culture in addition to the self-assessments described in the report? Does the CNSC require an external safety culture assessment when symptoms of licensee declining safety performance are detected?	All Canadian NPPs will either conduct periodic external (independent) assessments of safety culture or work in conjunction with an external safety culture expert to lead an assessment. Safety culture is an element of the WANO peer reviews. The CNSC has the authority under the NSCA to require such a review. The latest external assessment of safety culture was ordered by the Commission in 2009, in response to identified management deficiencies at a Canadian NPP. However, at this time the CNSC is focusing on developing a regulatory document clarifying roles and

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					expectations on safety culture.
74	Spain	Article 10	102	Does CNSC conduct periodic internal and external safety culture assessment? Has CSNC carried out any safety culture assessment?	Yes, the CNSC conducts internal safety culture assessments and its most recent was made in 2012.
75	United Kingdom	Article 10	10b	CNSC provides oversight of licensees' safety culture self-assessment programmes and processes; what steps does CNSC take to measure its own safety culture in order to demonstrate that CNSC makes nuclear safety the priority in all its activities?	The CNSC conducted an assessment of its safety culture most recently in 2012 which included the identification of those elements of safety culture that already exist at the CNSC as well as a phased-in approach to ensure continuous improvement in this area.

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ARTICLE 11: FINANCIAL AND HUMAN RESOURCES					
76	Spain	Article 11.1	106	According to the report, the Government announced its intention to bring forward in the fall of 2013 a new legislation that will update and enhance Canada's nuclear liability regime. Has it been presented to the Parliament?	On January 30, 2014 Bill C 22 the Energy Safety and Security Act was introduced in the Canadian Parliament. The Bill includes the Nuclear Liability and Compensation Act (NLCA), a modern nuclear liability regime which will replace the existing Nuclear Liability Act when adopted. Among the provisions of the NLCA will be raising the compensation limit to CDN $1 billion and enabling Canada to ratify the international Convention on Supplementary Compensation for Nuclear Damage (CSC). The CSC is a multilateral instrument which provides for additional compensation in the event of a nuclear incident. Canada signed the CSC on December 3, 2013. Bill C 22 must be reviewed by Parliament before it can be passed into law, following which Canadian ratification of the CSC will be permissible.
77	Spain	Article 11.1	114	According to the Annex 19 (iv), the program to develop Severe Accident Management Guidelines for CANDU reactors started in 2002 and concluded in early 2007. In other countries this program started more than ten years earlier and was fully implemented in the nineties. It seems that there was not a very proactive reaction in the CANDU reactors and industry in	The design on Canadian plants have always considered events which usually were not part of design basis on other jurisdictions, such as large break loss of coolant with loss of emergency core cooling. Correspondingly, the operators put in place emergency operating procedures to deal with events involving core damage since the beginning of the plant operation. The design provisions and operating procedures in place were judged to be adequate to deal with events similar to the Three Mile

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				relation with this subject. This is a relevant matter because Canada has to lead the countries that operate CANDU reactors. Did the industry develop appropriated severe accident research programs after TMI and Chernobyl accidents to support a straightforward development of SAMG? Did the regulatory body promote and require appropriated measure in a timely manner? Do you think that the current research funds and programs, from both the industry and the regulatory body, are sufficient to cope with the safety challenges posed by the Fukushima accident, including human factors under extreme conditions, in a time scale commensurate with its safety significance?	Island accident. However, the formalized set of guidelines addressing specifically large scale fuel meltdown was developed in the time frame of 2002-2007. These guidelines build on the international experience and research results. Fukushima lessons learned are applied to further enhance the existing SAMG, for example by explicitly addressing the multi-unit considerations.
78	France	Article 11.2	§ 11.2 - p.107 to 115	How does Canada ensure that there are sufficient and competent employees available in contractors' and subcontractors' staff to carry out any tasks important for safety, in particular during outages?	Having sufficient qualified staff to safely operate the licensed facilities is a requirement of the Canadian Regulations. This is met through Systematic Approach to Training (SAT) based training programs for operations, maintenance and other staffs that carry out safety related work. For outage based work, the majority of that work is carried out by the facilities base staff.

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					Where supplemental staff is required, licensees have agreements with the trades unions to provide staff. These staff will take supplemental training before the start of the outage to ensure that they are versed in the requirements and safety culture of the nuclear facility.
79	Korea, Republic of	Article 11.2	106	According to the INFCIRC/572/Rev.4 of "Guidelines regarding national reports under the convention on nuclear safety", it would be appropriate to include descriptions about plant simulators and their uses for training. Please describe the status and activities of simulator training, capabilities of plant simulators with regard to fidelity to the each plant and scope of simulation. Please explain how the CNSC ensures that simulator fidelity could be maintained and updated to the current plant conditions incorporating design modifications.	CNSC requirements for simulators at NPPs are documented in Regulatory Document RD-204 Certification of Persons Working at Nuclear Power Plants. RD-204 is reference in all Canadian Power Reactor Operating Licences and the relevant parts are repeated here: The licensee shall ensure that each NPP has in service a full scope simulator facility for training and examining persons seeking or holding a certification as reactor operator, unit 0 operator, control room shift supervisor or plant shift supervisor. The simulator shall be capable of simulating, realistically and in real time, all significant NPP maneuvers and transients that may occur under normal and abnormal operating conditions, including: • NPP start-ups and shutdowns; • Major NPP upsets and accident conditions; and • All significant failures of systems and their equipment and the consequences of such failures.

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					For conditions and failures that may vary in magnitude, such as pipe breaks, loss of inventory, loss of flow, loss of pressure, and loss of vacuum, the simulator shall have adjustable rates to simulate all possible degrees of severity of a condition or failure that impact on unit response or operator actions. A more complete list of simulation capabilities is included in CNSC Examination Guide EG2: Requirements and Guidelines for Simulator-based Certification Examinations for Shift Personnel at Nuclear Power Plants, which is also referenced in all Canadian Power Reactor Operating Licences. This list is included in Appendix 1 of RD-204. The CNSC expects licensees to establish and maintain processes and procedures that assure simulator fidelity. In general, licensees treat the simulator as an additional unit, so that when changes are made in the field or control room, needed changes to the simulator are made at the same time. Licensees have imbedded these processes and procedures in their engineering change control processes and procedures to ensure the work processes are seamless. Licensees regularly review simulator performance against actual plant performance and adjust as necessary. Where possible, licensees validate proposed changes to procedures or NPP systems in the simulator

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					prior to implementation in the NPP. CNSC Staff also review simulator fidelity during training and certification examination compliance inspections. The Full-Scope Simulators support the Initial and Continuing Certification Training Programs for the Certified Staff. The original scope of simulation for these simulators was restricted to the suite of Design Basis Accidents initiated from the Full Power Steady State. Over the past two decades, the scope of simulation has expanded to include an increasing number of tasks from the respective Job and Task Analysis associated with the Certified Operational Positions. Licensees are, on their own initiative, updating simulation capabilities to span a wider range of operating states. Licensees ensure simulator fidelity through the programmatic links from the Engineering Change Control process and the Operations Documentation Revision process into the Certification Training Programs. In addition to extensive performance testing on all simulation software releases, the simulators are calibrated to significant Operational Events that occur in the respective generating stations. At least one licensee conducts quarterly Simulator Review Board meetings attended by staff from the Simulator Support, Authorization Training and Examination Section as well as representatives from

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<tr><td></td><td></td><td></td><td></td><td></td><td>Operations. The meetings review simulator performance, emerging issues and schedules for training, maintenance and upgrades.</td></tr>
<tr><td>80</td><td>Russian Federation</td><td>Article 11.2</td><td>Section 11.2</td><td>What is the average staff schedule of Canadian NPPs, as well as the personnel of supporting companies?</td><td>The sense of the question is unclear; however, work hours vary greatly depending on duties:
Days based operations and maintenance staff typically works 40 hour work weeks.
Facility support staff (engineers, training instructors, managers, contractors, etc.) typically work 35 to 40 hour work weeks.
Shift operations staff works an average of 40 hours per week but in practice an actual work week varies from 36 hours to 48 hours dependent on the shift schedule (Bruce Power and OPG use a 5 crew shift schedule while NB Power uses a 6 crew shift schedule).</td></tr>
<tr><td colspan="6">ARTICLE 12: HUMAN FACTORS</td></tr>
<tr><td>81</td><td>India</td><td>Article 12</td><td>Section-12(j) Page 122</td><td>The following is stated in the report:-
‘In addition to identifying closure criteria for the actions and reviewing the submissions from Licensees, CNSC staff are engaged in other multi-faceted deliverables to ensure that the safety of Canada’s Nuclear facilities is enhanced in light of the</td><td>(i) The CNSC does not plan to develop one specific guide to address Fukushima HOF lessons learned. Instead, consistent with its integrated approach the CNSC has taken its HOF-related Fukushima response and lessons learned and incorporated these across the CNSC’s regulatory framework.
Since 2011, revisions to key regulatory documents have been developed, or are in progress, which contain elements</td></tr>
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				lessons learned from Fukushima on human and organizational factors. These deliverables will focus on: • Ensuring the lessons on human and organizational factors learned from Fukushima are incorporated in the new and revised elements of the CNSC's regulatory framework. • Pursuing research to establish a better understanding of decision making in severe, unanticipated situations.' Can Canada please provide following:- (i) Is it planned to bring out a safety guide for addressing human and organizational factors based on Fukushima experience. (ii) What are the elements and scope of research on decision making in severe, unanticipated situations?	corresponding with HOF lessons learned. These include proposed revisions to the "General Nuclear Safety and Control Regulations" to address human performance and fitness for duty (see DIS-13-02, http://www.nuclearsafety.gc.ca/eng/acts-and-regulations/consultation/comment/d-13-02.cfm); revisions to RD-353 "Testing the Implementation of Emergency Measures"; RD-2.10.1 "Nuclear Emergency Preparedness and Response"; and G-306 "Severe Accident Management Programs for Nuclear Reactors. In addition, HOF-related lessons learned have been incorporated into revisions of regulatory documents related to new builds such as RD-337 "Design of New Nuclear Power Plants", RD-360 "Life Extension of Nuclear Power Plants", and RD-369 "Licence Application Guide: Licence to Construct a Nuclear Power Plant". The CNSC also intends to include considerations related to BDBAs in its upcoming revision to G-323 "Ensuring the Presence of Sufficient Qualified Staff at Class I Nuclear Facilities - Minimum Staff Complement". (ii) At this time the CNSC is participating in the NEA/CSNI/WGHOF initiative on the topic of "human performance under extreme conditions", which considers human factors, organizational factors and infrastructure. Work

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					is planned that relates to decision-making in severe, unanticipated situations, which includes new control room systems, including decision-aiding and electronic procedures. Literature reviews are planned to address human reactions to severe accident conditions and human factors engineering approaches to design for severe accident conditions.
82	Korea, Republic of	Article 12	page 116	According to the description of page 116, the CNSC performs several activities to address human and organizational factors. Please explain how many staffs with human factors expertise are working in the utility and the regulatory body, and what kinds of duties they perform in their organization.	Regulator: The CNSC HOF group is composed of 12 HOF specialists at different seniority levels and competencies in different areas of sub specialties (e.g., human factors in design, organizational aspects, safety culture, etc). Key responsibilities include: • Leading the development/ maintenance of CNSC's regulatory framework • Analyzing and assessing licensee submissions pertaining to HOF; • Leading or participating in technical licensing and compliance work; • Preparing recommendations, reports and other documentation; • Reviewing and analyzing licensee events; • Managing projects related to an area of specialization; • Representing the CNSC at national and

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					international technical/ scientific meetings and organizations; and • Training and coaching staff members and external colleagues. Licensees: There is some degree of variance with respect to the level of human factors expertise across the industry. As an example, across OPG, there are a large number of people working in this broader “human factors/human performance” area. All OPG Engineering staff attend a Conduct of Engineering half day workshop at least once a year. In the narrower area of Human Factors Engineering, OPG has five HFE specialists, plus OPG makes use of additional HFE specialists from several external engineering service providers. The HFE specialists duties center on design activities embedded in the engineering change control process, but other non-design tasks are also undertaken such as: • minimum staff complement assessments, • reviews of procedure effectiveness, evaluation of Emergency Response Organization exercise response effectiveness,

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					• assessment of equipment and display fitness for purpose, participation in COMS (constructability, operability, maintainability, and safety) and other system walkdowns, etc.
83	Spain	Article 12	121	Does CNSC have any program to oversee the licensee organizational changes and how the decisions on organizational changes are taken, justified, documented and communicated? Are there any circumstances in which any particular organizational change has to be approved by CNSC?	For an operating licence, CNSC requires licensees to submit documentation which describes the Management System to be implemented. This documentation is reviewed against the requirements of CSA standard for Management Systems N286, which has a requirement to define the organization structure, responsibilities of management positions and interfaces internal and external. The licensee is required to notify the CNSC of any changes made to this documentation. In addition, the licensee is required to report annually the organization changes made in that year. This annual report addresses positions to a lower level than that of the management positions in the Management System documentation. The CNSC evaluation process verifies that the licensee used a comprehensive and systematic process to arrive at a safety-oriented organization and the basis for any change is rational and supported by clear records (e.g. documentation, a systematic assessment, outcome measures). In addition, the organization and subsequent changes must comply with key regulatory requirements,

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					those stated in CSA standard N286, minimum shift complement and hours of work. The CNSC approves and certifies individuals for a number of positions within the organization, however does not approve organization structures and subsequent changes.
84	Spain	Article 12	118	In relation with the human performance improvement, are CNSC or the licensees carrying out any research or development activities on human and organizational behavior under severe stress and extreme situations?	At this time the CNSC is participating in the NEA/CSNI/WGHOF initiative on the topic of “human performance under extreme conditions”, which considers human factors, organizational factors and infrastructure. Work is planned that relates to decision-making in severe, unanticipated situations, which includes new control room systems, including decision-aiding and electronic procedures. Literature reviews are planned to address human reactions to severe accident conditions and human factors engineering approaches to design for severe accident conditions. There is no R&D work being completed by licensees at this time.
85	Spain	Article 12	pg. 48 and 284	A sustainable operations plan has been developed to address the challenges associated with approaching the end of Pickering B commercial operation. According to the report, the changes and plans deal primarily with people-related issues and	In terms of maintaining a highly motivated work force, the plan focuses on the following areas: • Employee communications • Nuclear Safety Culture • Labour Relations

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				business issues pertaining to the life expectancy of the NPP. Could you provide additional information on the people-related issues and the actions planned to maintain a highly motivated staff when approaching the end of operational life?	• Staffing & Succession Planning • Retirements, Attrition and New Hires • Authorized staff plans • Employee engagement • Human performance plans • Industry interface strategy
ARTICLE 13: QUALITY ASSURANCE					
86	Korea, Republic of	Article 13	(a), page 123	It is mentioned in Article 13 that "The Class I Nuclear Facilities Regulations require licence applicants to propose their quality assurance (QA) programs for the site preparation, construction, operation, and decommissioning activities to be licensed, and in 13(a) that Licences for the activities to be licensed also include, directly or indirectly, the following QA / management system standards. .... ASME NQA-1". What measures are used for CNSC to monitor the quality related activities of licensee's contractors and to verify their conformity to ASME NQA-1?	The management system requirements for the purchasing of materials and services, as described in CSA N286-05, outline the conditions under which contractors are to be managed and their work verified against the requirements set by the licensee. This would include conformance to any code or standard set as a condition in the purchasing documentation for a contractor.

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87	Romania	Article 13	13 (b)	What specific regulatory inspection activities are performed to verify the measures taken by the licensees for preventing intrusion of counterfeit, fraudulent and suspect items into the nuclear supply chain?	Primarily, the CSA N286 standard requirement that licensees must have programs in place to ensure that "all procured items and materials meet the technical and regulatory requirements necessary for its use" applies. (A counterfeit or fraudulent item, by definition, will not meet technical requirements, unless you intentionally went out to purchase a counterfeit or fraudulent item.) This is why it falls under our procurement inspections. Other N286 requirements that also apply are: • Work activities shall be (b) carried out using approved … (ii) materials; (iii) parts; (iv) tools; • Designs, documents, tools, materials, parts, processes, services, and practices that do not meet requirements shall be identified and recorded as problems Purchasing, receiving, storage, issuance, and return of material, equipment, and services shall be controlled and shall include: • confirmation of the traceability of material in accordance with applicable codes, standards, and specifications; • confirmation that received material continues to meet requirements The selection of a supplier shall be based, in part, on an evaluation of the supplier's ability to deliver a technically acceptable product or

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					service. The evaluation shall confirm that the products or services meet technical requirements, including safety, reliability, and maintainability. Inspection and verification shall be planned, documented, and carried out by the organization responsible to ensure that items and/or services meet the requirements of the contract With respect to the procurement inspections, or any other inspections, we have not carried out any to date that explicitly focused on CFSI. However, the CNSC has recently added CFSI as one of the criteria in its Type II inspection guides, which will be used in upcoming inspections. Following a recent presentation to the Commission, CNSC staff plan to carry out more focused oversight of the licensee's program to prevent and detect CFSI.
88	Spain	Article 13	Pg. 123	Could you explain the equivalence between the standard CSA N286-12 “Management system requirements for nuclear power plants” and the IAEA document “The management system for facilities and activities” (GS-R-3)?	The CSA management system standard N286-12 entitled “Management system requirements for nuclear facilities” is a further evolution of the CSA N286-05 standard “Management System requirements for nuclear power plants”. N286-12, as in GS-R-3, is applicable to all Nuclear facilities and has as a first principle “safety is the paramount consideration guiding decisions and actions”. It improves on GS-R-3 by outlining some

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					specific requirements for various types of licensees.
89	Spain	Article 13	Pg. 125	When will the management systems of all Canadian NPP be in accordance with the new CSA N286-12 standard?	CSA N286-12 was published in 2012. It will be referenced in the Licence Condition Handbook for all NPP Power Reactor Operating Licence renewals starting in 2014.
90	United Kingdom	Article 13	Management Systems	Please provide information on licensees' quality assurance processes and CNSC's oversight arrangements of these to provide assurance of the safety and security of the supply chain providing components and personnel providing a safety role in the Canadian nuclear industry.	The CNSC does oversight of the licensees supply management processes to ensure they continue to meet the requirements of the CSA N286 standard. "Supply Management" is the focus of one of the guides used for inspections that are part of the CNSC's baseline plan. CNSC staff also does routine reviews of supply management documentation as a desktop activity. The requirements in CSA N286 apply to both supply of materials and services; therefore both components and personnel involved in safety related system activities are subject of the oversight activities.
ARTICLE 14: ASSESMENT AND VERIFICATION OF SAFETY					
91	China	Article 14.1	P 11	Based on the Appendix B of this report, the four units of Pickering B have operated for 30 years by the end of 2015, which means their PT's life time is well beyond the CANDU6 PT's. In addition to that, they still can keep operating to 2020. Could you please give us some key explanation about the	It is not planned to replace the Pickering B pressure tubes as part of the limited life extension of the units. While the "assumed design life" of the pressure tubes is 30 yrs (210,000 Effective Full Power Hours (EFPH), or approximately end calendar year 2015, extensive analysis and laboratory testing has been completed to demonstrate that the pressure tubes are fit for service for at least 247,000 EFPHs. The testing and analysis have

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				justification process?	been submitted to the regulator and the rational for operation beyond the assumed design life has been accepted. A full ISR for the Pickering B units was completed, along with an Environment Assessment and a Global Assessment. The results of these assessments formed the basis for the Integrated Implementation Plan (IIP), which defined the inputs to the Continued Operations Plan (COP). The COP is the living document that describes all of the work that must be done to life extend the station to at least 2020. As the Pickering B station enters its life extension phase at the end of 2015, all of the COP actions will be completed by this time.
92	France	Article 14.1	Annex 14 (i) (d) - p. 281	Canada indicates that some licensees have conducted a “PSA-based seismic margin assessment”, producing results such as the seismic capacity of the NPPs. In this method, a 0.3 g review-level earthquake is specified for most plants east of the Rocky Mountains. Could Canada specify if CNSC intend to request sensitivity studies regarding the review-level earthquake to detect cliff-edge effects?	Sensitivity studies have been part of the discussion between industry and CNSC staff on how to address the issue of cliff-edge effects. The issue is still on-going.
93	Germany	Article 14.1	page 141	Recognizing that an ISR provides	Under the current version of RD-360, the ISR

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				an opportunity to re-evaluate the entire safety case for an NPP, the CNSC Fukushima Task Force considered that PSRs should be done regularly for all NPPs. A 10-year frequency, in line with international practice, was judged reasonable and capable of being integrated into the licensing process. The CNSC Action Plan assigned an action to the CNSC to consider the development of a regulatory framework for the implementation of the PSR process. CNSC staff has proposed that any ISR conducted for an NPP should be considered the first PSR for that NPP. The CNSC is planning to update RD-360 to focus on periodic performance of ISR, to be conducted in conjunction with licence renewal. The results of such ISR/PSRs, summarized in an integrated implementation plan, would become part of the licensing basis for the NPP. As required by CNSC the Integrated Safety Review (ISR) has to be performed each time an application for a licence renewal is	is conducted once approximately every 30 years in support of continued operation. Once revised, the licence and RD-360 will require licensees to perform the same type of review every 10 years. The PSR approach would generate much of the information needed in support of licence applications, so this information would be prepared as part of the PSR and then submitted to CNSC in support of licence renewal. With a systematic approach to addressing continued fitness for service in place, the maturity of the compliance program, and the Commissions ability to revoke a licence or stop operation at any point if safety is compromised, a proposal to increase licence periods to 10 years will be tabled in conjunction with the implementation of PSR.

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				submitted to the regulatory body. Typically, this is necessary every five years. A periodic safety review (PSR) has to be performed every 10 years. It would be appreciated, if Canada could elaborate in more detail on the differences between the ISR and the PSR and the implications on the licence renewal considering the different frequencies of both.	
94	India	Article 14.1	Section-14(i) (d) Page 136	The following is stated in the report: "The Licensees are enhancing models for beyond design basis accidents to align with the requirements of S-294 and are analyzing them systematically – focusing on multi unit events, irradiated fuel being events and accidents triggered by extreme external events". The CNS task force has also recommended enhancements of models of beyond-design-basis accidents, including ones developed for multi units. Can Canada please clarify; (i) What is exactly meant by enhancement of models for	(i) "Enhancement of models for beyond design basis accidents" refers to the Probabilistic Safety Assessment (PSA) models enhanced with refined assumptions. These models are developed by reducing the conservatism in the baseline PSA through focused thermal hydraulic analyses to determine the environmental conditions, such as those in the powerhouse, and to support more realistic modeling for these scenarios. For example in the baseline Level 2 PSA model, any sequences that might result in severe core damage at two or more units are conservatively assigned the consequence of a four unit scenario. In the enhanced model new MAAP-CANDU analyses were performed to better assess the consequences of scenarios leading to severe core damage in two and four units. In addition to crediting new analyses, the

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				beyond design basis accidents? (ii) How are multi-unit events taken in to consideration in PSA studies? (iii) In a multi-unit site, in which way accident progression is considered in different units initiated from a common external event.	enhanced model assesses the benefits from the safety improvement opportunities which credits the safety improvements such as the addition of Filtered Containment Venting System, the addition of portable emergency generators, and the provision of an alternate and independent supply of water as an emergency heat sink providing make-up water to the heat transport system. (ii) PSA study reflects a single reference unit modeled in detail. The PSA study is extended to include events that can affect more than one unit. The events can affect the unit by initiating a process transient in the reference unit and/or by affecting the reliability of shared mitigating systems. For example: • A Steam Line Break in adjacent unit can initiate a process transient on the reference unit and can affect the reliability of common mitigating systems • A Loss of Offsite Power (LOOP) initiates a process transient on the reference unit and affects the reliability of common mitigating systems (iii) External events such as seismic events were treated as fully correlated events. That is, the seismic event was assumed to affect all four units in exactly the same manner at exactly the same time. Therefore, for external events Severe Core Damage (SCD) leads

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					directly to a Large Release (SCD Frequency = LR Frequency). LRF is both a unit and a site metric. In the future other mitigating actions such as SAMG actions involving use of portable equipments will need to be modelled, and the safety goals for a multiunit site will need to be defined.
95	Korea, Republic of	Article 14.1	c, 128	What is the limit of containment hydrogen control in design basis accident and severe accident in relation to Fukushima follow-up?	The limiting DBA for hydrogen control is the Large LOCA with ECC impairment. Note that this event would be considered a BDBA outside of Canada and may be reclassified under revisions to CNSC regulatory documents. The limiting DBA is unchanged by the Fukushima follow-up. Note that, as a follow-up action to Fukushima, the installation of passive auto-catalytic recombiners was accelerated for stations where it had not already been completed. Hydrogen source terms in severe accidents are potentially higher than for DBAs, particularly if core-concrete interaction is postulated. An integrated approach to prevention and mitigation is being adopted by licensees. Such an approach would use the features of the CANDU design to prevent or arrest core damage at the earliest possible stage. This strategy aims at preventing core damage or limiting it to fuel clad oxidation inside fuel channels, or retaining molten core inside the

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					calandria vessel. As part of this strategy, containment venting is under consideration. Filtered containment venting has already been installed at Point Lepreau and is planned during the refurbishment of Darlington. Options are still being evaluated for Bruce. Pickering is scheduled to close in 2020 and a major upgrade is not likely.
96	Korea, Republic of	Article 14.1	d, 133	1. What is the legal basis of the 3 year PSA update cycle? Why is the update cycle extended to 5 year in S-294 revision? 2. PSA results are used in SAMG development. What are the mitigation strategies used in regulatory agencies review process and full power and shutdown / low power SAMG about CANDU reactor SAMG.	1. Licensees have a requirement in their license to comply with the S-294 standard which required them to update the PSA every three years. The three year period was taken from Section 4.2 of IAEA-TECDOC-1106, which states: “Modifications that impact the PSA results may require an immediate updating of the LPSA. However, even if this type of modification does not arise for a longer period, it is still suggested that the updating process be audited every three years and the LPSA formally amended at that time”. The S-294 standard is being revised to align the PSA update with the safety analysis report update and with the license renewal. 2. PSA insights are one of the inputs in developing the Severe Accident Management Guidelines as well as in considering potential design modifications to lower frequency of certain events identified by PSA studies. CNSC, in its reviews of PSA and SAMG, uses both the guidance developed by the IAEA as

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					well as the Canadian regulatory documents, such as S-294 and G-306. These documents are being revised to include Fukushima lessons learned.
97	Romania	Article 14.1	p.132	It is mentioned on page 132 that "in response to Fukushima, the NPP licensees have performed or are planning to perform deterministic analyses for representative severe core damage accidents. Such safety analysis has already been conducted as part of the ISR to decide on the scope of refurbishment activity for NPPs undergoing life extension. The licensees are enhancing their models for beyond-design basis accidents to specifically address multi-unit events." What are the regulatory guidelines and criteria used by CNSC staff for reviewing severe accident analyses performed by the licensees?	CNSC benefits from the guidance provided in the IAEA documents as well as Canadian regulatory documents such as RD-310 and G-306. The overarching criteria applied in the evaluation of severe accidents are the Safety Goals which establish targets for frequencies of core damage and releases of fission products. It must be acknowledged that the practices in this area are undergoing rapid development both in response to the Fukushima event and due to the emerging data from research activities.
98	Spain	Article 14.1	Pg. 138	Could you provide some information on the experience of applying a WANO corporate peer review to AECL, a non NPP facility?	Through WANO corporate peer review, AECL has benefited from improved corporate alignment and oversight on nuclear safety.

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99	Spain	Article 14.1	141	In order to incorporate the Periodic Safety Review (PSR) in the regulatory framework, CNSC staff has proposed that any Integrate Safety Review (ISR) conducted for an NPP should be considered the first PSR for that NPP and CNSC is planning to update RD-360 Life extension of Nuclear Power Plants to focus on periodic performance of ISR, to be conducted in conjunction with license renewal. This way of integrating the PSR concept in the regulatory frame work could be reasonable for the current operating plants, but future plants will be excluded of PRS until they reach the life extension period. How could you regulate the realization of PSR for new reactor every ten years before reaching the life extension period?	For existing plants, adopting the ISR as the first PSR is a reasonable approach as the ISR process was modeled after the PSR, has the same outputs and is focused on verifying the validity of fitness for service for continued operation. For a newly built plant, the first licence issued will require the licensee to conduct a PSR due 10 years after initial fuel loading in accordance with the successor to RD-360. There is no need to wait for the ISR to be considered the first PSR, as it will be deemed to be a PSR.
100	Spain	Article 14.1	Appendix G, pg. 227	Could you provide some updated information on the situation of the LBLOCA-related Category-3 safety issues? According to the report the resolution of those safety issues were expected by the end of 2013.	Licensees have completed a joint project on Large Break LOCA and submitted the project closure report and supporting documents to CNSC for review. Licensees requested: • CNSC consent for use of the Composite Analytical Approach (CAA) for future licensing safety analyses of LBLOCAs. • Re-categorization of 3 LBLOCA-related

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					CSIs from category 3 to category 2 based on the work completed in this joint project. A decision by CNSC is expected by late spring 2014.
101	Spain	Article 14.1	Appendix G, pg. 227	Does CNSC have any international forum of experts to discuss the Category-3 safety issues? Has CNSC considered inviting an international peer review for the Category-3 safety issues, the action plan established and the schedule for resolution?	For some category 3 CSI, an independent technical panel has been invited to review the issue, such as CSI PF20 (Analysis methodology for NOP/ROP trips) and PF18 (Fuel bundle/element behaviour under post dry-out conditions). The independent reviews have been completed and industry/CNSC are reviewing/working on the results
102	Switzerland	Article 14.1	p.130	The effectiveness of the safety systems shall be such that for any serious process failure: o the exposure of any individual of the population shall not exceed 5 mSv o the exposure of the population at risk shall not exceed 100 person-Sv For any postulated combination of a (single) process failure and failure of a safety system (dual failure), the predicted dose to any individual shall not exceed 250 mSv to the whole body or 2.5 Sv to the thyroid. The IAEA Guidelines SSG-2 /	The accident classification and corresponding dose limits quoted from page 130 are those in force at original licensing and pre-date IAEA SSG-2 by many decades. The classification into single failures and dual failures is approximately frequency based but does not map exactly onto more modern, frequency-based schemes. In 2008 CNSC published RD-310, Safety Analysis for Nuclear Power Plants that classifies accidents by frequency as: • AOO (f > 10-2/y) • DBA (10-2 > f > 10-5/y) • BDBA (f < 10-5/y) Licensees are currently implementing safety analysis that meets the requirements of RD-

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				Article 2.9 specifically requires that the frequency of an accident be referred to in groupin accidents. Table 2 in this document refers to specific probability levels. Yet the passage defines doses for serious process failure referring to categories and frequency of occurence. Could the probability of these serious process failures been given and groupe according to the IAEA recommendation? What would be the corresponding radiological doses?	310. When completed, the analysis will be fully in line with IAEA SSG-2. For new NPPs, the dose acceptance criteria are given in RD-337, Design of New Nuclear Power Plants, and are 0.5mSv for AOO and 20mSv for DBA. For existing NPPs, the dose acceptance criteria can be regarded as targets. Safety goals are defined in RD-337 for BDBA.
103	United Kingdom	Article 14.1	14 (i) (h)	Please identify what additional requirements for safety assessment will be placed on licensees as a result of the adoption of a PSR process over and above those currently required by the operating licence renewal process.	The current relicensing review focuses on both the licensee’s performance history over the last licensing period and the adequacy of licensee programs as well as future plans for the next licence period. A PSR would enhance this framework by requiring a review where the licensee is to reassess the actual state and operation of the plant against the current licensing basis and then against modern codes and standards, including those for design.
104	United States of America	Article 14.1	Pages 78 and 142	Please clarify if the CNSC procedures and tools for RIDM (including document Q850) incorporate concepts of IAEA standards for risk application such	Yes, appendix A, specifically A.4.2.2 of the CNSC RIDM basis document describes the application of PSA into the RIDM process, addressing (at the time draft versions of) SSG-3 and SSG-4.

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				as SSG3 and SSG4?	
105	China	Article 14.2	P 144	The fuel channel lifecycle management is particularly important for the safe and economic operation of CANDU NPP. Could you please give some detailed introduction of the latest result of this The fuel channel lifecycle management project?	A well-established pressure tube periodic material surveillance program satisfying the requirements of CSA Standard N285.4-05 regularly removes pressure tubes for examination and testing of key material properties, including fracture toughness. To date all fracture toughness measurements from ex-service (surveillance) pressure tubes have been above the CSA N285.8-10 lower bound fracture toughness curve, thus meeting the CSA N285.4-05 acceptance criteria for fracture toughness measurements. In recognition of the potential effect of higher Hydrogen equivalent level [Heq] on pressure tube fracture toughness as [Heq] levels increase with increasing reactor service hours, the industry established the Fuel Channel Life Management Project (FCLMP) beginning in 2010. The Fuel Channel Life Management Project was designed to gather physical evidence of changes with increasing [Heq], update the requirements of CSA N285.8 as appropriate, and develop new approaches to core assessments. As part of the FCLMP, fracture toughness measurements have been obtained on ex-service pressure tube material artificially hydrided to [Heq] levels ranging from 63 to 124 ppm:

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					• Fracture toughness values gained from the tests conducted at normal operating temperatures (≥ 250°C) are consistent with prior fracture toughness results, i.e. above the CSA N285.8-10 lower bound fracture toughness curve • Some burst test results of hydrided ex-service material at transition temperatures (< 250 °C, to address reactor heat-up and cool-down conditions) have shown fracture toughness results below the CSA N285.8-10 fracture toughness curve indicating the need to develop new fracture toughness models to account for hydrogen effect on fracture toughness behavior in low and transition temperature regions. The FCLMP work developed new pressure tube fracture toughness models that address the effect of [Heq] and chlorine concentrations. The development of these new models followed the principles of CSA N285.8-10 Clause 8 for updating of material property models and include: • Cohesive-Zone based fracture toughness model , for transition temperatures (< 250°C); and • Statistical fracture toughness model for normal operating temperature (≥ 250°C). Both models have been submitted to the CNSC and subjected to independent third-party expert

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					review. The third party review has been conducted with full transparency and engagement of CNSC specialists in the review process.
106	Finland	Article 14.2	chapter 14 (ii)	What is the typical number of staff of permanent inspectors at NPP sites? What is the training and competence of the permanent inspectors to be able to assess the importance different phenomena and findings (e.g. mechanical, NDE, I&C, civil engineering, system expertise etc.)?	The on-site inspection staffing levels in 2013 were: • Single Unit Station Gentilly 2: 4 • Single unit station Point Lepreau: 4; • Multi Unit (8 reactors) Bruce Site: 8; • Multi Unit (8 reactors) Pickering: 11; • Multi Unit (4 Reactors) Darlington: 9; The Inspector Training and Qualification Programme entail the development and implementation of an effective, standardized and systematic approach for training and qualifying all CNSC inspectors. The program is composed of a combination of core training, service-line specific training and on-the-job training. If a technical matter needs to be evaluated by an inspector that is outside their specific expertise a team of experts, or specialists are available for consultation within the CNSC.
107	Finland	Article 14.2	Ch. 14 (ii) a	The periodic inspection programs may include non destructive in-service inspections (ISI) of the safety important SSC's; What are the requirements for	(1) Following the methodology of the European Network of Inspection Qualification (ENIQ), the Inspection Qualification (IQ) process was established in 2008 to meet the requirements of Canadian Standards

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				qualification of the NDE inspections (methodology and personnel) and how the regulator controls the conformity of the ISI processes? In other words, how it is ensured that possible defects initiating and growing in the mechanical components during operation can be detected with a high reliability? To what extent risk based approach is applied in selection of the ISI inspection targets (RI-ISI)?	Association (CSA) Standard N285.4-05. With concurrence of the Canadian nuclear regulator, the CANDU Inspection Qualification Bureau (CIQB) was founded in 2009 to provide an independent Third Party Review and qualification of critical and complex in-service inspections (ISI). For each of the CANDU Structures, Systems and Components (SSCs) subject to periodic inspection per the CSA N285.4 Standard (Fuel Channels, Steam Generators, Feeders, and Piping) an Inspection Specification was issued that details the plausible degradation mechanisms and the inspection requirements for each mechanism (i.e. measurement and sizing accuracy, probability of detection, etc.). Within these SSC's, more than 40 individual inspection methodologies were selected for qualification through the CIQB. This qualification process reviews: i) the Inspection Procedure (IP), ii) the technical justification (TJ) which defends the procedure, and iii) the Training and Qualification Plan (TQP) for inspection personnel, all against the recognized Inspection Specification. A successful CIQB review results in a certificate that qualifies that specific IP. Annual IQ Update meetings formally report status and progress to the regulator. The CIQB conducts annual internal audits, and the

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					regulator conducts regular and random audits of selected licensees. (2) Currently there is no licence/regulatory requirement in Canada to follow risk based/risk informed approach in selection of periodic or in-service inspection locations. Canadian utilities follow Canadian Standards Association (CSA) N285.4 and CSA N285.5 Standards which are based on deterministic rules to define the inspection scope. Given the international trend to move to Risk Informed In-Service Inspection (RI-ISI), CANDU industry has been involved in determining the path forward on application of RI-ISI on CANDU plants since 2009. A Candu Owners group (COG) RI-ISI pilot study on Darlington Unit 2 (with focus on primary side systems) was competed in 2011 with following objectives: • Develop CANDU best fit RI-ISI methodology • Implement the methodology on several selected systems • Perform delta risk assessment (CSA N285.4 vs. RI-ISI) • Explore risk reduction opportunities The Project concluded that: • CSA N285.4 has an implicit risk-related rationale

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					• EPRI RI-ISI methodology can be adapted to the CANDU design • There is negligible delta risk in moving from CSA N285.4 to RI-ISI • CSA N285.4 includes conservatism in the Periodic Inspection Program scope Consensus was later reached by industry that new CSA N285.7 Standard, "Periodic Inspection of CANDU Nuclear Power Plan Balance of Plant Systems and Components", is the logical first step for the incorporation of RI-ISI into the CANDU licensing basis. CSA N285.7 Standard which fully adapts RI-ISI methodology is under development and is scheduled for publication for end 2015 according to CSA Master Schedule. A second COG RI-ISI pilot study on Darlington Unit 2 balance of plant systems and components is being performed to support the development of CSA N285.7 Standard.
108	Finland	Article 14.2	Ch. 14 (ii) b	Annex 14 (ii) (b) summarizes e.g. Flow Accelerated Corrosion Program. What kind of pipe thickness measurement program is used to support the software (CHECWORKS) analysis before and during operating? Annex 14 (ii) (b) describes also component replacement program from other plants or	1) As required by licence conditions, all Canadian NPPs are required to have in-service inspection programs for safety related balance of plant systems as well as periodic inspection programs for nuclear systems. Unless otherwise specified, the inspections conducted in support of the CHECWORKS software are completed using ultrasonic thickness techniques. Radiography is used for piping which has socket weld fitting or is less than

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				manufacturers' storehouses. How it is confirmed that the NPP's have sufficient amount of spare parts available for abnormal situations when emergency cooling and other auxiliary systems would be needed for a longer time?	7.62cm in diameter. Material analysis for chromium content is also included in the inspection. 2) Identification of critical spare parts is part of the aging management programs at Canadian NPPs. To ensure these parts are available, even in abnormal situations, inventories are kept a level that is sufficient for the needs of the NPPs. The Emergency Spares Assistance Process is also available should the need arise and has been used in the past to procure parts to maintain maintenance outage critical paths (using both Canadian and United States sources). Canadian NPPs have also installed upgrades as a result of the Fukushima event that would be available to maintain reactor heat sinks if required.
109	Japan	Article 14.2	14(ii)(b), p144	Canadian report describes approaches for aging management, such as, the life extension and operation beyond the original design lives. At the time of life extension, will any special inspections be conducted for safety significant components and structures? If yes, what kind of inspection will be conducted for each critical component and structure?	One of the major elements to assess the safety operation of the plant for the extended period is the "Integrated Safety Review" (ISR). The ISR is a comprehensive assessment of plant safety performed by the licensee in accordance with IAEA Safety Guide on Periodic Safety Review (PSR) of Nuclear Power Plants [SSG-25]. This includes condition assessment (CA) of these SSCs and reviews of the effectiveness of SSC specific ageing management programs. The results of these reviews should establish for each SSC subject to ageing review whether there is a need for any special inspections to confirm understanding of aging behaviour and

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					actual condition of SSC as well as to evaluate if any additional inspection and/or aging management practices will be required for long term operation. The special inspections typically are required for areas that cannot be inspected during a normal maintenance outage, and/or are not covered by existing Periodic Inspection or In-Service inspection programs. For example, one of the typical CA recommendations of particular safety significance is the inspection of the internals of the calandria vessel (CV) and the reactor control units (RCU) components. These are performed during the refurbishment outage to confirm structural integrity of the CV for continued operation, as the removal of the fuel channel assemblies and feeders provides a one time opportunity to conduct a detailed internal inspection of the CV since its construction. Other special inspections identified by the ISR/CA process might include normally inaccessible areas of the containment and reactor vault structures, end shields, dousing tanks, buried piping, as typical examples. Before the unit is returned to service, a leakage rate test is also required for the containment structures to demonstrate their leak tightness.
ARTICLE 15: RADIATION PROTECTION					
110	Argentina	Article 15	Article 15; Section	Is there some regulatory	As defined in the Class 1 Nuclear Facilities

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			15 (a); pages 146	requirement in Canada about maintaining an exclusion area around a NPP? If yes, please, could you give details about the criteria used to define the exclusion area? (Article 15; Section 15 (a); pages 146 / 147)	Regulations: An exclusion zone is: "A parcel of land within or surrounding a nuclear facility on which there is no permanent dwelling and over which a licensee has the legal authority to exercise control." Section 3 of the Class 1 Nuclear Facilities Regulations requires an applicant for a licence (i.e. for a Nuclear Reactor facility) to provide "a description of the site of the activity to be licensed, including the location of any exclusion zone and any structures within that zone" It is important to interpret the wording of this regulation carefully. The way it is written means that an applicant could propose that no exclusion zone is needed (or interpreted otherwise: a 0m exclusion zone). This has been used for very small research reactors such as the SLOWPOKE designs. An exclusion zone is measured as a radius from the outer wall of each reactor building. There is no fixed exclusion zone size requirement in Canada although most nuclear power plants currently have an exclusion zone radius of 914m (or 1000 yards) based on original proposals made in the 1960s. The applicant is expected to propose and defend the extent of the exclusion zone based on a series of considerations discussed below. An exclusion zone is established based on several factors including (without being

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					limited to): • Land usage needs (i.e. how much land the project will require) • The performance of the design during normal operation or as a result of accidents and malfunctions whether generated internally or from a combination of external events (malevolent or naturally occurring) and anticipated doses at the exclusion zone boundary. • Emergency Preparedness (onsite and offsite) including evacuation needs • Environmental Factors (e.g. projected wind strength and direction) • Security and robustness. (e.g. how secure is the plant against threats and how robust is the design itself?) Radiological Dose Criteria: • The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. • Under normal operating conditions, the effective dose at the exclusion zone boundary to a person who is not a nuclear energy worker does not exceed 1 mSv over

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					the period of one calendar year; • Under Anticipated Operational Occurrence (AOO) conditions, the effective dose at the exclusion zone boundary to a person who is not a nuclear energy worker does not exceed 0.5 mSv over the release time due to the AOO • Under Design Basis Accident (DBA) conditions, the effective dose at the exclusion zone boundary to a person who is not a nuclear energy worker does not exceed 20 mSv over the release time due to the DBA.
111	Czech Republic	Article 15	Section 15(a)/Page 146	“In addition, section 13 of the Radiation Protection Regulations requires that every licensee 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 • 1 mSv per year for a person who is not a nuclear energy worker (i.e. the public)” Is the 4 mSv limit valid also for	The fetus in Canada does not have legal status; the 4 mSv applies to the mother only, but is considered to be protective of the fetus.

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				the fetus of the pregnant nuclear energy worker?	
112	France	Article 15	§ 15 (b) - p.149	Considering the ALARA principle for public exposure, does Canada intend to update the practice of using the public exposure limit (1 mSv) to set the limits of releases for normal operation?	Canada is updating the practice of using the public exposure limit (1mSv) to set up the release limits for normal operation. The approach consists of developing a process using a dose criterion, instead of 1 mSv, based on performance of a specific nuclear sub-sector as a whole. The release limit would then be back-calculated using the Environmental Transfer Model to calculate the dose to a member of the public (representative person). This is in accordance with the Canadian Standard Association Guidelines CSA N288.1-08. Ref: Canadian Standard Association Guidelines CSA N288.1- 08. Guidelines for calculating derived release limits for radioactive material in airborne and liquid effluents for normal operation of nuclear.
113	Germany	Article 15	Page 148, Section 15 (b)	It is mentioned that various measures are in place to reduce doses to workers from exposure to tritium and to train workers on potential tritium hazards, which is recognized as good practice. Some licensees have dehumidifiers on the air inlets of reactor buildings and/or alarming area tritium monitors. Some licensees also de-	Point Lepreau has dehumidifiers on the reactor building air inlets. Both OPG and Bruce Power use portable dehumidifiers during maintenance outages in the reactor vault/building to reduce tritium exposure. These are just some of a number of examples of measures currently employed by NPP licensees to keep tritium exposures ALARA, although not prescriptively mandated by the CNSC through the Nuclear Safety and Control

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				tritiate their heavy-water inventory. As this was already stated in the last report, has the number of licensees who implemented these measures increased since then, or are there plans to generally adopt these measures to all licensees?	Act and associated Regulations. However, ALARA is a regulatory requirement and CNSC staff conducts regular compliance verification activities to ensure that all licensees are implementing the ALARA principle through the use of the best available technologies and methods to ensure that radiation exposures and doses to persons are kept ALARA.
114	India	Article 15	Table F.3, Page 222 of Appendix F	As per the table on performance ratings of NPPs, in the year 2010, Bruce A was under below expectation (BE) category in the area of radiation protection. And in the year 2011 and 2012, Bruce A was shifted to category Satisfactory (SA). Please elaborate the actions taken that have resulted in overall performance improvement.	The actions taken that resulted in the improved rating for Bruce Power included: • Performance of a risk identification and characterization, • Implementation of work controls, and • Enhancements to their alpha Radiation Protection Program.
115	Ireland	Article 15	p 150	It is noted that dietary and behavioral habits, age and metabolism are taken into account when assessing the doses to the 'members of the public with the greatest exposure'. How are the characteristics of the aboriginal population (in particular in relation to the diet) taken account, e.g. habit surveys data?	Dietary surveys are used to obtain food ingestion data specific to aboriginal communities. When specific information is not available, data from similar communities may be used.

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116	Korea, Republic of	Article 15	4.2, 146	1. In ICRP publication 60, once pregnancy is declared, the conceptus should be protected through the application of a supplementary equivalent dose limit to the surface of the woman's abdomen (lower trunk) of 2 mSv for the remainder of the pregnancy and through the limitation of intakes of radionuclides to about 1/20 of the Annual Limit of Intake (ALI). However, dose limit to a pregnant nuclear energy worker is set as 4 mSv. What is the basis for 4 mSv? 2. Please explain dose limits (or risk limits) and assessment point to protect the public in case of normal operation, DBA and severe accidents.	1. In Canada, after extensive consultation, it was determined that 4 mSv for the balance of the pregnancy, once declared, was the most appropriate annual effective dose limit for a pregnant nuclear energy worker. This dose limit was deemed to be practical, yet adequately protective and would not limit work opportunities for women in the nuclear industry. During consultation, many female members of the workforce stated that they felt that a limit of 1 mSv for the balance of a pregnancy (as per ICRP publication 60) was too restrictive. The concern was raised that this limit could discourage employment opportunities for females and favour hiring males. 2. Normal operations public dose limits are provided in the Canadian Nuclear Safety Commission’s Radiation Protection Regulations as follows: • Effective dose: 1 mSv per year • Equivalent dose to the lens of an eye: 15 mSv per year • Equivalent dose to the skin: 50 mSv per year • Equivalent dose to the hands and feet: 50 mSv per year During the control of an emergency and the consequent immediate and urgent remedial

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					work the effective dose limit is 500 mSv and the equivalent dose limit to the skin shall not exceed 5000 mSv. These emergency dose limits do not apply to nuclear energy workers who have informed their employers that they are pregnant. Note that changes to these requirements are currently being proposed via amendments to the Radiation Protection Regulations. However, broader protection of the public during an emergency is handled primarily by the municipality and province in which the emergency is occurring. The federal government may lead and/or coordinate if the emergency has impacts on a more national level or originates outside the country. Intervention guidelines in terms of taking protective actions are outlined in the respective provincial and federal emergency response plans.
117	Pakistan	Article 15	Section 15(a), Page 146	Canada may please explain how the Radon doses are distinguished from the occupational exposures.	Occupational Radon doses (e.g., uranium mining) are measured mainly by personal alpha dosimeters (PAD). The dosimeter meets technical and quality assurance requirements outlined in a CNSC standard (S-106 Rev.1) and is offered by a licensed dosimetry service. In some cases, each worker wears his or her own PAD. In other cases, a PAD is worn by one individual and the dose measured is applied to several members of a group of people who were working close to one another under similar conditions. Finally, another option involves estimation by utilizing air

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					monitoring results combined with time/occupancy information. Exposure to radon progeny are calculated as Working Level Months and the information is submitted to the National Dose Registry. In work places where radon exposures are not associated with the licensed activity itself, radon doses are generally not measured. This exposure is considered background/baseline and thus is not subject to the dose limits. Note that Health Canada does have guidelines for residential radon exposure (for example).
118	Pakistan	Article 15	Section 15(a), Page 146	It is stated that dose limit to a pregnant worker is 4mSv during the course of her pregnancy and dose limit to a member of public is 1 mSv/year. Canada may like to share the basis of establishing the dose limits for pregnant worker?	In Canada, after extensive consultation, it was determined that 4 mSv for the balance of the pregnancy, once declared, was the most appropriate annual effective dose limit for a pregnant nuclear energy worker. This dose limit was deemed to be practical, yet adequately protective. During consultation, many female members of the workforce stated that they felt that a limit of1 mSv for the balance of a pregnancy (as per ICRP publication 60) was too restrictive. The concern was raised that this limit could discourage employment opportunities for females and favour hiring males.
119	Pakistan	Article 15	Section 15(b), Page 148	The individual doses at NPP "Bruce A and B" and NPP "Pickering A and B" seem to be on higher side and generally show	Caution should be used when comparing the collective effective dose data between Canadian NPPs; such a comparison is not entirely appropriate, due to the differences

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				increasing trend, if compared with other NPPs. Canada may please elaborate the causes of higher individual doses at these NPPs ?	between individual stations (such as design, age, operation and maintenance). At Bruce A and Bruce B, the collective effective doses from 2010 – 2012 are attributed primarily to the large scope of work required during planned maintenance outages for life extension and equipment lifecycle engineering plans, as well as the refurbishment activities at Bruce A Units 1 and 2, which were completed in 2012. At Pickering A and B, the collective effective doses from 2010 - 2012 are primarily due to the number and scope of outages, and the extensive outage programs and modifications executed during these planned outages to improve operations and ensure safe and reliable performance to the end of commercial operation. Some forced outages also contributed to this dose trend.
120	Spain	Article 15	Pg. 149	Could CNSC confirm that all the recommendations made by the Radiation Safety Institute of Canada in relation with the alpha contamination incident in the Bruce A NPP has been adequately address and are closed?	CNSC staff has reviewed the Radiation Safety Institute of Canada report and have taken action to ensure that Bruce Power addresses the recommendations cited in the report. Bruce Power has made many improvements to their Radiation Protection Program, particularly in the area of alpha monitoring and control. The CNSC continues to expect that Bruce Power will take all measures necessary to implement an effective radiation protection program and CNSC staff ensure this is the case

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					through routine compliance verification activities.
121	Switzerland	Article 15	n.a.	The effective dose limit for occupational exposure of the personal should not exceed 50 mSv. Will this value soon be reduced to 20 mSv as in the most countries?	The CNSC is undertaking amendments to sections of the Radiation Protection Regulations. These amendments will harmonize Regulations with updated international standards, and also clarify requirements and address gaps identified in light of the nuclear incident at the Fukushima Daiichi nuclear power plant in Japan. No changes are proposed to the effective dose limits in Canada for occupational exposures of Nuclear Energy Workers, which are 50 mSv in a one-year period and 100 mSv over a five-year dosimetry period. The effective dose limits in Canada are consistent with recommendations of the International Commission on Radiological Protection (ICRP) Publication 103 and the recommendations of the IAEA's Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards – General Safety Requirements, Part 3, (Interim Edition), 2011.

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ARTICLE 16: EMERGENCY PREPAREDNESS					
122	China	Article 16.1	P162	Has Point Lepreau completed the code of source term estimation? Is it a real-time code or an unreal-time one?	As the Canadian national report identifies, Point Lepreau does not perform source term estimation in support of offsite emergency response. However, Point Lepreau has recently committed to the Canadian Nuclear Safety Commission that it will develop the capability and explore the use of the Emergency Response Projection (ERP) code. The results of the code would be used by a Technical Advisory Group supporting the provincial New Brunswick Emergency Measures Organization for off-site emergency response and decision support.
123	Czech Republic	Article 16.1	Page 163	How often is the Federal Nuclear emergency Plan (FNEP) updated?	The most recent FNEP revision was started in 2011 and completed in 2013. The previous revision was done in 2002. There is no formal established revision period, but rather the initiation of a revision is driven by changes in the operating environment (e.g. updated IAEA guidance) or operational experience and lessons learned (e.g. Fukushima).
124	Czech Republic	Article 16.1	Page 166	Is there any plan for organizing future full-scale exercises?	Federal organizations responsible for the Federal Nuclear Emergency Plan led by Health Canada and the Federal Emergency Response Plan led by Public Safety Canada along with stakeholders will hold a full-scale exercise in May 2014. Future exercises will be considered during the after-action reviews of this major exercise.

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125	France	Article 16.1	§ 16.1 (d) - p. 162	During an emergency response, the offsite authorities get a source term from the impacted licensee. Could Canada specify the measures taken to guarantee that the source term will be effectively delivered on time, especially during a multi-unit accident? How does Canada intend to set up arrangements which will provide independent assessment of an ongoing accident, in order to protect the surrounding population with other information than from the licensee source term?	Arrangements exist between the Operator and the Provinces for prompt notification of an event at NPPs and this includes obtaining station parameters such as the source term as soon as it becomes available. The development of source term data and its transmittal to offsite decision makers is tested regularly by all operators and is verified during CNSC inspections of emergency exercises. Provincial Nuclear Emergency Plans in Canada include default protective actions (e.g. evacuate 0-3 km and shelter 3-10 km) appropriate to the event classification (e.g. General Emergency) which will be initiated by the Province within 15 minutes being notified by the licensee of an emergency. Source term estimates are not required to initiate this response. In the Province of Ontario, source term is calculated using a site specific Emergency Response Projection code. The Province and licensee independently run the applicable Emergency Response Projection code (Bruce ERP, Darlington ERP or Pickering ERP) in parallel. Licensee Technical Support staff within the licensee Emergency Response Organization (ERO) communicate projection results to the Province and provide support to resolve any discrepancies thereby ensuring the Province has the best information available for decision making on protective actions for the public. ERP codes are being updated to address multi-unit events as part of

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					the Fukushima action plan. The province of New Brunswick is striving for greater alignment with the Ontario utilities in terms of adopting the use of ERP. The results of the code would be used by a Technical Advisory Group support the provincial New Brunswick Emergency Measures Organization for off-site emergency response and decision support.
126	France	Article 16.1	16.1(a), 154	The report states that one issue was the discrepancy between offsite versus onsite emergency preparedness; it states also that there are no significant gaps in emergency preparedness, at NPP, municipal, provincial or federal level. Could Canada clarify whether or not there is an issue in emergency preparedness?	The report makes reference to a discrepancy related specifically to the oversight of emergency preparedness, in particular the duality that exists in the Canadian approach whereby the licensee's emergency preparedness is formally and regularly evaluated by the regulator (CNSC), whereas there is no such formal and independent approach to evaluate the preparedness of offsite authorities. The level of preparedness of offsite authorities is self-managed by each authority, as well as in multi-lateral coordinating committees at the provincial and federal level.
127	Germany	Article 16.1	Page 300, Annex 16.1 (d)	In the Province of Quebec, an information campaign on nuclear-related risks took place in January 2012, in parallel with the distribution of new potassium iodine pills to residents and workers in the urgent protective	The authority and decision making for KI management lies with Provincial authorities. The Provinces of Québec and New Brunswick opted for a pre-distribution strategy whereby all households in their primary zones have instant access to KI pills. The Province of Ontario has KI available for residents to obtain

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				action planning zone within an 8-km radius around Gentilly-2. Please provide some more details on the general strategy for this measure? What is the situation for the other NPP's.	at their will before an emergency, and distributed in reception centres during an emergency. The strategy in Ontario is currently under review. Additional information on the Québec strategy is available at their website: http://www.urgencenucleaire.qc.ca/
128	Hungary	Article 16.1	Page 162, Chapter 16.1(d)	What kind of software is used for the source term estimation?	Ontario Power Generation and Bruce Power use site specific Emergency Response Projection codes that were developed by Ontario Hydro. These codes take information on plant radiation fields, metrological data and off site radiation monitoring into account in the source term estimation. New Brunswick Power is in the process of developing a similar software application for Point Lepreau as part of the CNSC Integrated Action Plan on the lessons learned from the Fukushima nuclear accident.
129	Hungary	Article 16.1	Page 166, Chapter 16.1(f)	How does CNSC check the emergency exercises? (Do they read the evaluation of the exercises or do they observe the exercise on-site?)	The CNSC has an elaborate inspection program which includes annual evaluations of a licensee's emergency exercises on-site. In addition, CNSC verifies the licensee's interactions with offsite stakeholders directly.
130	Hungary	Article 16.1	Page 166, Chapter 16.1(f)	How often does CNSC observe emergency exercises?	Emergency exercises are formally inspected annually. Additional smaller scale inspections may be done on particular aspects of the licensee's EP program (for example: observing medical response, fire response or offshift drills)

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131	India	Article 16.1	Sub-article 16.1 (c) Page 160	In the report, the CNSC requirements and expectations on emergency preparedness procedures for new build projects are detailed. It would be more appropriate if the size of exclusion zone and protective zone are mentioned. Does CNSC specify the exposure limit for emergency workers?	As defined in the Class 1 Nuclear Facilities Regulations: An exclusion zone is: "A parcel of land within or surrounding a nuclear facility on which there is no permanent dwelling and over which a licensee has the legal authority to exercise control." Section 3 of the Class 1 Nuclear Facilities Regulations requires an applicant for a licence (i.e. for a Nuclear Reactor facility) to provide "a description of the site of the activity to be licensed, including the location of any exclusion zone and any structures within that zone" It is important to interpret the wording of this regulation carefully. The way it is written means that an applicant could propose that no exclusion zone is needed (or interpreted otherwise: a 0m exclusion zone). This has been used for very small research reactors such as the SLOWPOKE designs. An exclusion zone is measured as a radius from the outer wall of each reactor building. There is no fixed exclusion zone size requirement in Canada although most nuclear power plants currently have an exclusion zone radius of 914m (or 1000 yards) based on original proposals made in the 1960s. The applicant is expected to propose and defend the extent of the exclusion zone based on a series of considerations discussed below.

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					An exclusion zone is established based on several factors including (without being limited to): • Land usage needs (i.e. how much land the project will require) • The performance of the design during normal operation or as a result of accidents and malfunctions whether generated internally or from a combination of external events (malevolent or naturally occurring) and anticipated doses at the exclusion zone boundary. • Emergency Preparedness (onsite and offsite) including evacuation needs • Environmental Factors (e.g. projected wind strength and direction) • Security and robustness. (e.g. how secure is the plant against threats and how robust is the design itself?) Radiological Dose Criteria: • The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. • Under normal operating conditions, the effective dose at the exclusion zone

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					boundary to a person who is not a nuclear energy worker does not exceed 1 mSv over the period of one calendar year; • Under Anticipated Operational Occurrence (AOO) conditions, the effective dose at the exclusion zone boundary to a person who is not a nuclear energy worker does not exceed 0.5 mSv over the release time due to the AOO • Under Design Basis Accident (DBA) conditions, the effective dose at the exclusion zone boundary to a person who is not a nuclear energy worker does not exceed 20 mSv over the release time due to the DBA. The size of protective zones is under the jurisdiction of the Provincial governments. The Provincial governments generally base their zone sizes on accepted international practice and IAEA guidance. CNSC radiation protection regulations include provision for emergency workers. These regulations are currently under review.
132	India	Article 16.1	Page 162	As per CNSC action plan, all the licensees were assigned a responsibility to install a real-time NPP boundary radiation monitoring system with an appropriate backup power and communication system. What is	A real-time radiation monitoring system is in place at three of the five NPP sites in Canada. The automated near-boundary gamma monitoring system for Darlington and Pickering nuclear generating stations were put in service in September 2012. These monitors are located approximately 1 km from the plant

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				the implementation status of this system? Is the system part of decision support system for emergency conditions? Please provide additional information with respect to this system and its usage.	to provide immediate information on dose rates at the site boundary. They are solar powered with an 8-hour battery backup capacity. The remaining two sites have committed to install such a system and are in various phases of planning and implementation. There are multiple benefits to such a system, for example: • assuring the public of absence of a release, • confirming the direction and magnitude of the release, • informing responder of potential contaminated or high dose rate areas. Emergency Management Ontario (EMO), the Ontario provincial emergency management authority, has endorsed this methodology as an enhancement to the current process of dispatching staff to complete gamma survey. Gentilly-2 nuclear generating station, which ceased operations on December 28th, 2012, has the same automated near-boundary gamma monitoring capability. This capability is being reevaluated as the station is going through main decommissioning activities for the next few years.
133	Japan	Article 16.1	16.1(a), p152	Canadian report says, "In Canada licensees of nuclear facilities are responsible for onsite emergency	The Canadian legal and regulatory framework clearly places the responsibility of onsite response with the operator/licensee. As the

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				planning, preparedness and response.” What organization in Federal government is responsible for those? Who oversees licensee’s onsite emergency planning, preparedness and response? On the other hand who oversees offsite emergency planning, preparedness and response?	Canadian nuclear regulator, the CNSC’s licensing and compliance processes verify that the licensee has adequate provisions to respond to an emergency by reviewing their emergency plans, preparedness programs and performance during exercises and drills. Oversight of offsite planning, preparedness and response is managed provincially via coordinating committees of all agencies involved. This is generally led by the provincial emergency management agency. A similar process is utilized at the federal level.
134	Spain	Article 16.1	158	According to the report, the CNSC Action Plan assigned an action to the licensees to evaluate and revise their emergency plans with regard to multi-unit accidents and severe external events. This activity was to include an assessment of their staff requirements to ensure their emergency response organizations are capable of responding effectively to multi-unit accidents or to severe natural disaster events. All NPPs submitted their assessments to the CNSC. We would appreciate further information on this subject and its results, specifically to	The approach used to deal with the human resources in severe accident had to be modified slightly. In Design basis accident, for a specific accident, circumstances are better defined and procedures are also well structured allowing the identification of knowledge and skills necessary. This is basically the approach prescribed in G-323 in order to determine the MSC which then forms a requirement at all time. Verification and validation is performed to confirm that the event response is indeed appropriate with the specified resources and qualifications. In BDBA, this approach had to be modified due to the uncertainties caused by the event itself. The Canadian approach was modified. For BDBA at this time we are using the concept of "sufficient number of qualified staff" to make a clear distinction between DBA

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				clarify whether an increase of the human resources available on site has been identified as a result of this review. Has the G-323 “Ensuring the Presence of Sufficient Qualified Staff at Class I Nuclear Facilities – Minimum Staff Complement” been used in this exercise?	requirements and BDBA which is not yet an established requirement. Licensees have developed and are validating new BDBA procedures. Based on this work, licensees will develop guidelines which would include details about resources needed to perform the emergency response tasks. These guidelines will be designed to assist the Emergency Organization in taking decisions on the tasks that could be accomplished with the available resources at the time. This concept is evolving as progress is being made. At this time, there have not been changes made to the Minimum Shift Complement level. The CNSC has an initiative to review the G-323 document and has yet to decide if the new concept for BDBA will be captured in this review.
135	Spain	Article 16.1	Pg. 159	Has CNSC analyzed the advantage that a new emergency building, similar the one available at Fukushima Daiichi, could imply to centralize and facilitate the on-site emergency management and activities in case of severe accident with significant radioactive releases? Does the Canadians NPP plan to erect such emergency facilities?	Generally, CNSC uses a more performance-based approach to regulation. The requirements for emergency facilities are currently being updated via the development of regulatory requirements for licensee emergency preparedness (REGDOC 2.10.1 “Emergency Management and Fire Protection: Nuclear Emergency Preparedness and Response” (http://www.nuclearsafety.gc.ca/pubs_catalogue/uploads/REGDOC-2-10-1-Emergency-Preparedness-Programs.pdf).

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					The revised requirements include provisions for licenses to consider the robustness (physical and radiation) of their emergency facilities.
136	Switzerland	Article 16.1	p.162	The report states that NPP licensees are working on improvements in areas of support to offsite emergency preparedness: source term estimation, plume modelling, radiation monitoring and dose modelling. Whose radiological assessment is taken as basis for the recommendation on protective actions, like sheltering or the intake of potassium iodide pills, the one provided by the licensee or the one of the offsite authority? Whose radiological assessment is taken as basis for the recommendation on protective actions, like sheltering or the intake of potassium iodide pills, the one provided by the licensee or the one of the offsite authority?	The authority for protective action lies clearly with the provincial governments. During an emergency, the province will obtain the station parameters from the operator and will review/analyze these to determine protective actions. In Ontario, the operator and the province utilize the same modeling codes. The province can consult with other organizations such as Health Canada and the CNSC to validate their calculations and planned protective actions
137	Switzerland	Article 16.1	p.162	The report states that the use of automated real-time field monitoring at an NPP boundary is a best practice. The CNSC Action Plan assigned an action to	The currently installed and planned systems are based on wireless data transmission with backup power. The stations are of a robust design, but are not specifically designed to withstand severe external events.

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				licensees to install automated real-time NPP boundary radiation monitoring systems with appropriate backup power and communications systems. Does the communication of the radiation monitoring system at the boundary or around an NPP support a wireless data transfer, and is it designed to withstand an external event, e.g., an earthquake?	
138	United Kingdom	Article 16.1	16.1 (f)	Article 16 provides information detailing CNSC's involvement in evaluating the 3 yearly full-scale emergency exercises, please provide information on the role of CNSC in evaluating smaller scale emergency exercises that might, for example, test the on-site capabilities including demonstration of out of hours emergency responder call out arrangements, multi unit event arrangements etc.	Emergency exercises are formally inspected annually. Additional smaller scale inspections may be done on particular aspects of the licensee's EP program (for example: observing medical response, fire response or offshift drills)
139	Switzerland	Article 16.2	p.168	The report states that an alerting system, coupled with the instructional messages broadcast over the radio and the television, will ensure that the population	In Ontario, the siren system functions on battery and is thus independent from the domestic electrical supply. The public alerting systems (sirens) have back up power which would be available for

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				within the primary zone (10 km) is notified appropriately and in a timely manner. In case of an accident due, e.g., to an external event, where the domestic electricity supply is unavailable, it is questionable whether the public is timely informed. How is the population within the primary zone notified in a timely manner, should the domestic electricity supply be unavailable due to an external event?	notifying people within 3 km. Public telephone systems do have back up power systems, however, if homes only have cordless phone systems timely alerting could be an issue with 10km. The Provincial EMO has the responsibility for notifying the public in the primary zone. In the event of a power outage, Police assistance would be sought to ensure this occurs in a timely manner. In New Brunswick, which has a lower density of residents, a pre-established door-to-door notification program is in place.
ARTICLE 17: SITING					
140	France	Article 17.1	§ 17(i) - p.174	Could Canada clarify how an external hazard can be eliminated from a site-specific safety assessment? Does the screening approach distinguish natural from human-induced external hazards in terms of elimination criteria?	In answer to the question 1, there are 7 screening criteria that are used: 1. The event is of lesser or equal damage potential than the events for which the plant has been designed. 2. The event has a significantly lower mean frequency of occurrence than another event that has been screened, and the event could not result in worse consequences than the other screened event. 3. The event cannot occur at the site or close enough to the site to affect the plant. 4. The event is included in the definition of another event.

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					5. The event is slow in developing such that it can be demonstrated that there is sufficient time to eliminate the source of the threat or provide an adequate response. 6. The event does not cause an initiating event (including the need for a controlled shutdown) and safety system function loss(es) needed for the event. 7. The consequences to the plant do not require the actuation of front-line systems In response to question 2, the screening process does not distinguish natural from human-induced external hazards in terms of elimination criteria.
141	Germany	Article 17.1	page 174	Licensees also have to perform a site-specific external hazards screening to identify other hazards that may require a PSA or a bounding analysis. Further, the licensees must consider combinations of events, including consequential and correlated events. Examples of consequential events include external events (such as a cooling water intake blockage caused by severe weather or a tsunami caused by an earthquake) and internal events (such as a fire caused by an earthquake). Examples of	For coincidental independent events, the licensees may follow the accepted screening criteria and screen out the coincidental events if the frequency is lower than the frequency screening level.

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				correlated events include heavy rainfall concurrent with a storm surge or high winds caused by a hurricane. Consequential events are considered in the PSAs (see subsection 14(i)(d)). Selected cases are documented in the NPP safety reports (see subsection 14(i)(c)). Canada improved its approach to address site specific external hazards by considering correlated events as well as consequential events. Can Canada discuss if independent combination of events, where its simultaneous occurrence has to be assumed due to their relative high frequency and degree of damage needs to be addressed by the licences, too?	
142	Ireland	Article 17.1	Article 17 and 18, p 171-186	Regulatory framework and licensing process for New Build: are applicants required by law (or encouraged) to assess trans-boundary effects of potential severe accidents scenarios (in the Environmental impact Statement, Siting (Art. 17) and Design (Art.	Applicants are required to assess the effects of potential severe accidents scenarios. Regulatory document RD-346, “Site Evaluation for New Nuclear Power Plants”, states that prior to construction, the proponent confirms with the surrounding municipalities and the affected provinces, territories, foreign states, and neighbouring countries, that

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				18) requirements)?	implementation of their respective emergency plans and related protective actions will not be compromised for the life cycle of the proposed site.
143	Spain	Article 17.1	Pg. 178	The CNSC Integrated Action Plan required all major nuclear facility licensees to complete the review of the basis for external events against modern state-of-the art practices. According to the report, for NPPs that have not been refurbished, the magnitudes of the external events considered in the designs comply with the standards applicable at the time of original licensing and are generally very conservative. However, the event magnitudes considered were below modern international best practice in some cases. Is CNSC considering requiring to those facilities a reassessment of the external events design bases according to state-of-the art and modern standards?	Yes. After the Fukushima accident review, CNSC included in the action plan, for licensees to conduct more comprehensive assessments of site-specific external hazards. CNSC's Fukushima task force issued two site specific action items: • FAI 2.1.1 - Re-evaluation, using modern calculations and state of the art methods, of the site specific magnitudes of each external event to which the plant may be susceptible. • FAI 2.1.2 - Evaluate if the current site specific design protection for each external event assessed in 1 above is sufficient. If gaps are identified a corrective plan should be proposed
144	Spain	Article 17.1	Pg. 178	The CNSC Fukushima Task Force recommended that the licensees should conduct more comprehensive assessments of site-specific external hazards to demonstrate that: a) considerations	These assessments have just recently been completed, and the licensees are still in the process of developing and implementing plans to address any potential weaknesses. Some actions have been completed or are in

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				of magnitudes of design-basis and beyond-design-basis external hazards are consistent with current best international practices and b) consequences of events triggered by external hazards are within applicable limits. The licensees have completed various tasks in response to this recommendation, including reviewing the bases of external events, completing or updating PSAs and expanding their application to analyze site-specific, external hazards. What actions have been taken or decided as a result of this review?	progress. For example: • Ontario Power Generation has installed flood protection barriers at locations that may be subjected to external flooding under severe weather conditions (eg, at standby and emergency power generator buildings at Darlington, and at the standby generator fuel forwarding building at Pickering 1-4). Ontario Power Generation is also pre-staging staff and aligning critical valves in support of severe high wind response at Pickering 5-8. • Bruce Power has started a campaign to reinforce fasteners (bolts) on building cladding for high wind events. The province of New Brunswick is still in the process of further evaluating site-specific external hazards for seismic, tsunami and high winds, which is expected to be completed by mid-2014. Following completion of the seismic hazard assessment, PSA-based Seismic Margin Assessment methodology will be examined to determine if any changes are warranted, and to identify any potential impact on existing equipment qualification, design guides, seismic analysis, etc. For tsunami and wind hazards, they will undergo screening in accordance with our procedures to determine if any further work to develop detailed PSA is warranted.

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145	Spain	Article 17.1	Annex 8, pg. 252	The CNSC requested an IAEA International Seismic Safety Centre "Site & External Events Design" (SEED) review service for all NPPs. A pre-SEED mission to define the scope of review is expected in summer–fall of 2013 and a full SEED mission is expected in late spring 2014. Has the pre-SEED mission been carried out? Which will be the scope and content of the SEED mission	In January 2013 the CNSC initiated informal discussions with the IAEA to determine the feasibility of a potential International Seismic Safety Centre "Site & External Events Design" (SEED) review service mission for all Canadian NPPs. At this time, CNSC staff is collecting industry feedback regarding the re-evaluation of site specific seismic hazards. Following completion of this re-evaluation phase, a pre-SEED mission to define the scope of a full SEED mission is expected to take place before the end calendar year 2014, following which formal undertakings will be initiated with the IAEA
146	Switzerland	Article 17.1	p.178	On Page 178, it is stated: The rationale for the magnitudes selected for beyond-design-basis hazards was not documented adequately and consistently for all the NPPs that were not refurbished. Further, the scope of the assessments and event magnitudes considered were below modern international best practice in some cases [...]." What does CNSC consider to be modern international best practice regarding event magnitudes of external hazards?	The following are considered modern international best practice: ASMI/ANS RA-Sa-2009, ANSI/ANS-58.21-2007, NS-R-3, NS-G-1.5, NS-G-3.1, NS-G-3.5, SSG3 Pub1430, CSA-289.1, CSA-290, and from the CNSC RD-310 "Safety Analysis for Nuclear Power Plants" and RD-337 "Design of New Nuclear Power Plants".
147	Argentina	Article 17.3	Article 17; Section	Can you give details about the original design for beyond-design-	There is no "original design for beyond-design-basis events" for existing plants for

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			17 (iii) (c); page 1	basis events that do not comply with modern practices? (Article 17; Section 17 (iii) (c); page 178)	external hazards. The concept for beyond design basis events related to external hazards, everywhere around the world, came some time after the plants went into operation and it was realized that, due to the uncertainty in the original design basis, those may actually be exceeded. More recently the standards for the design have been revised, for example in the Canadian CSA Standard (CSA N289.1) the Design Basis Earthquake probability of exceedance has been changed from a 1 x 10-3 per year (earlier publication in 1980) to 1x10-4 per year in its recent publication in 2008.
148	Japan	Article 17.3	17(iii)(a), p177	Canadian report says, “the licensees examined events more severe than those that have historically been regarded as credible.” How are these more severe events incorporated into the future postulated design –basis events? Are these licensees’ examinations just for Fukushima follow-up?	The licensee examinations were triggered as a result of Fukushima follow-up and lessons learned. There is no requirement or intent to incorporate these more severe events into the design basis. The hazard evaluation discussed in Article 17(iii)(a) showed that the design basis assumptions remain valid. Canada is currently developing guidance for design extension conditions (DEC), and some of these more severe events may be included within the guidance on DEC.
149	Euratom	Article 17.4	page 178	Do the arrangements referred to with the US allow for the two concerned parties to make their own, independent assessment of the likely safety impact on their own territory of a proposed nuclear installation?	The current arrangements referred to in the Report do not impact on either country’s ability to render independent decisions in the siting or construction of a nuclear power plant. Rather the close relationship that has evolved through formal and informal collaborative efforts over the past several years ensures that

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					both country's point of views are considered early in the decision process by the respective regulatory agency before a final decision is rendered.
ARTICLE 18: DESIGN AND CONSTRUCTION					
150	France	Article 18.1	§ 18 (i) - p.185	Could Canada briefly describe modifications which have been or will be set up (material or organizational, on-site or off-site, etc.) to improve abilities to withstand prolonged losses (heat sinks or power supply)?	CANDU NPPs have large inventories of water that can be used as passive heat sinks in various scenarios, including the loss of electrical power. The water available for passive cooling includes water in the secondary cooling system, the primary cooling system, the moderator and the calandria vault / shield tank. Canadian NPPs also have independent and diverse backup power supplies onsite with enough fuel for many days of emergency power generation. Since the Fukushima accident, a number of improvements have been made to provide additional make-up water and electrical power. This is based on portable equipment, otherwise known as emergency mitigating equipment, stored on-site or near the site. Details vary by station as there are significant differences between station designs and the practicability of modifications. For Bruce Power, emergency mitigating equipment, consisting of portable diesel pumps and generators to supplement the existing emergency and backup equipment, has been procured and deployed. Operational

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					procedures and guidelines for deployment of this equipment have been issued. These guidelines also provide validation procedures for training. Upgrades related to the evaluation of alternate coolant make-up to the reactor included the installation of external make-up lines and provision of additional relief capacity to the calandria vault. For example, Bruce Power reported that water connections to the steam generators in all Bruce units are complete. An assessment of the practicality for installing additional overpressure protection to the shield tank is underway at Bruce. For Darlington, emergency mitigating equipment work included the development of instructions and training, completion of storage buildings, and deployment of field runs. The equipment includes portable pumps, portable generators, hoses and connections, and personnel communication equipment stored onsite as well as additional equipment and resources stored offsite. A station emergency drill for Darlington was completed in August 2013, with deployment of emergency mitigation equipment, and a report on the drill was issued to validate instructions and timing. For Pickering, emergency mitigating equipment work included the development of instructions and training, completion of storage buildings, and deployment of field runs. The equipment includes portable pumps, portable

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					generators, hoses and connections, and personnel communication equipment stored onsite, as well as additional equipment and resources stored offsite. A station emergency drill for Pickering was completed in February 2013, with deployment of emergency mitigation equipment, and a report on the drill was issued to validate instructions and timing. For Point Lepreau, plans are underway, and detailed engineering is in progress for design changes related to the emergency mitigation equipment. NB Power has provided a plan and schedule for the evaluation of alternate coolant make-up to the reactor. Design upgrades include the installation of additional connections to the primary heat transport system, steam generators, and moderator system. The detailed engineering work is in progress, with installation expected during the next planned outage in Spring 2014.
151	Germany	Article 18.1	page 182/183	Vendor pre-project design reviews … The process is divided into three distinct phases… Can Canada elaborate in more detail on the three phases and the rational/objectives of each phase?	The reviews take place in three phases, each of which is conducted against related CNSC regulatory documents and Canadian codes & standards: Phase 1: Pre-Licensing Assessment of Compliance with Regulatory Requirements - This phase involves an overall assessment of the vendor’s nuclear power plant design against the most recent CNSC design requirements for new nuclear power plants in Canada as indicated in Design of New Nuclear

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					Power Plants (RD-337) or Design of Small Reactor Facilities (RD-367) as applicable, as well as all other related CNSC regulatory documents and Canadian codes & standards. Phase 2: Pre-Licensing Assessment for Any Potential Fundamental Barriers to Licensing - This phase goes into further details with a focus on identifying any potential fundamental barriers to licensing the vendor's nuclear power plant design in Canada. Phase 3 Follow-up - This phase allows the vendor to follow-up on certain aspects of Phase 2 findings by: • seeking more information from the CNSC about a Phase 2 topic; and/or • asking the CNSC to review activities taken by the vendor towards the reactor's design readiness, following the completion of Phase 2. For more Information on the CNSC's Pre-licensing Vendor Design Review, please refer to GD-385, Pre-licensing Review of a Vendor's Reactor Design.
152	India	Article 18.1	Annex 18(i) Page 319-320	Control facilities for personnel involved in accident management' is indicated as one of the elements for verifying effectiveness and supplementing wherever appropriate.	The control facilities include main control rooms and secondary control areas and are on-site. Licensees also identify a technical support centre which can provide technical advice to both accident management and to emergency management staff.

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				Can Canada indicate which, are these control facilities – are they separate from control rooms? If these control facilities are technical support centres – then are they on-site or off-site and what are their design features?	Fukushima Action Item 1.9.1 requires licensees to perform "An evaluation of the habitability of control facilities under conditions arising from beyond-design-basis and severe accidents." Licensees have worked together in a CANDU Owners Group joint project to develop a methodology for assessing habitability. Licensees also consider the performance of field actions in executing severe accident management guidelines. In addition to the control facilities referred to in Annex 18(i), licensees also have facilities for emergency management. Arrangements vary between stations but include a designated emergency management facility and a back-up facility. CNSC is developing REGDOC-2.10.1, Nuclear Emergency Preparedness and Response that includes requirements and guidance for emergency facilities. Note that these requirements do not become mandatory until the document is approved by the Commission and the document is included in the facility licence.
153	India	Article 18.1	18(i) Page-184, 318	During refurbishment of Bruce units-1 & 2, addition of more shutdown system trip parameters ……….. Etc. Kindly brief on the following points. 1. What are these trip parameters ? 2. Trip parameters added based on	1) A heat transport system (HTS) high pressure trip enhancement has been implemented in Units 3 and 4 and will be implemented in Units 1 and 2 as the core ages (within 10 years of return to service). The scope of the restart project included additional trips on Moderator high pressure (SDS1) and moderator low flow (SDS2), these were not installed due to the fact

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				Fukushima accident. 3. Change in the trip settings in shutdown system 2 on neutronic overpower and its basis.	that analysis showed that they would provide no additional safety benefit. The scope also included changes trip setpoints of SDS1 HTS low flow trip loop, Neutronic overpower (NOP) 3 pump operation hand switch trip and the High Log N Rate trip loop. Trip setpoints for boiler low and hi levels in SDS 1 and SDS2 were also changed due to the installation of new boilers. 2) There were no changes as a direct result of the Fukushima accident. 3) The trip settings for the SDS 2 Neutronic overpower (NOP) trips were not changed. What was done was the installation of additional SDS2 flux detectors into vertical guide tubes to enhance the overall NOP trip coverage.
154	India	Article 18.1	18 (i), Page-184, 321	It is mentioned that "The licensees are demonstrating that the equipment and instrumentation necessary for SAM and essential to the execution of SAMG will perform their function in the severe accident environment for the duration for which they are needed." Kindly elaborate such qualification requirements (environmental parameters and mission time) and their acceptance criteria by CNSC.	Fukushima Action Item 1.8.1 requires licensees to develop a plan and schedule for performing assessments of equipment and instrumentation survivability. Licensees have worked together in a CANDU Owners Group joint project to develop a methodology for equipment survivability, based on EPRI work. Application of the methodology is ongoing. Formal acceptance criteria have not been developed; however, CNSC has stated that it does not expect the assessments to be as rigorous as full equipment environmental

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					qualification. A new standard is being developed by the CSA Group that will address design and safety analysis requirements for Beyond Design Basis Accidents.
155	India	Article 18.1	18(i), Page-184, 320	Is there any regulatory acceptance criteria/ requirements being evolved by CNSC for containment filtered venting system (like radionuclide characterization and filter efficiency)? What is the time frame for implementation of such systems.	There are no specific acceptance criteria set by CNSC for containment filtered venting. The effectiveness can be judged against the overall safety goals established for NPPs. For new NPPs, the safety goals are established in CNSC RD-337, Design of New Nuclear Power Plants. For existing NPPs, individual licensees propose safety goals for their PRAs which are accepted by CNSC. Point Lepreau has already installed filtered containment venting (it was installed at refurbishment and pre-dates the Fukushima accident). The remaining Canadian NPPs have vacuum containments and already have Emergency Filtered Air Discharge Systems, but these are not qualified for severe accidents. Darlington has committed to installing additional containment venting during refurbishment, currently planned for 2016-2021. Bruce Power is still evaluating options. Pickering is planned for closure by 2020 and installation of additional containment venting may not be feasible.
156	India	Article 18.1	Annex. 18, Page-315	Annexure 18 of the report mentions about the design	CNSC is developing REGDOC-2.10.1, Nuclear Emergency Preparedness and

Ser	Country	Original Reference	Reference in Report	Questions/Comment	Response
				requirements and design assessments .Can Canada clarify that Emergency Support Centre is equipped to handle severe accident condition along with extreme external event affecting the plant site?	Response that includes requirements and guidance for emergency facilities. Note that these requirements do not become mandatory until the document is approved by the Commission and the document is included in the facility licence. Emergency Support Centres at Canadian NPPs are not all currently equipped to handle extreme external events. However, all NPPs have backup facilities that can be used if the primary emergency facility is lost.
157	Korea, Republic of	Article 18.1	182	In 182 page of Article 18, it is stated that "Those amendments will update selected design-basis and beyond-design-basis requirements and expectations". Please provide more detailed information on the amendments in the requirements for design-basis and beyond-design-basis.	CNSC reviewed and updated a number of regulatory documents, including those related to design basis and beyond design basis requirements: • RD-337, Design of New NPPs is being updated as REGDOC-2.5.2 • RD-310, Safety Analysis for NPPs is being updated as REGDOC-2.4.1 • S-294, Probabilistic Safety Assessment is being updated as REGDOC-2.4.2 These documents are expected to be approved in early 2014. Major changes to safety analysis were not found to be necessary. Some changes were made to highlight analysis for multiple units on a site and requirements to address cliff-edge effects. Some additional requirements for design of new NPPs were identified, in particular, those

Ser	Country	Original Reference	Reference in Report	Questions/Comment	Response
					related to: • provision of alternate coolant makeup • provision of alternate electrical power • requirements related to multiple units on a site A new standard is being developed by the CSA Group that will address design and safety analysis requirements for Beyond Design Basis Accidents.
158	Spain	Article 18.1	181	Explain if the regulatory document RD-337 includes requirements to achieve safety objectives equivalents to the following ones: a) new nuclear power plants have to be designed, sited, constructed, commissioned and operated with the objective, among others, of "enhancing the effectiveness of the independence between all levels of defence-in-depth, in particular through diversity provisions (in addition to the strengthening of each of these levels separately…), to provide as far as reasonably achievable, an overall reinforcement of defence-in-depth." b) The potential radioactive releases to the environment from	a) The concept of defence in depth shall be applied to all organizational, behavioural, and design related safety and security activities to ensure they are subject to overlapping provisions. The levels of defence in depth shall be independent to the extent practicable. If a failure were to occur, the defence in depth approach allows the failure to be detected, and to be compensated for or corrected. This concept shall be applied throughout the design process and operation of the plant to provide a series of levels of defence aimed at preventing accidents, and ensuring appropriate protection in the event that prevention fails. b) the dose limits and safety goals established in RD-337 (and the updated regulatory document REGDOC 2.5.2), along with the requirements for containment design and performance in RD-337 enable new designs in

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				accidents with core melt5, also in the long term6, should be minimized by following the qualitative criteria below: o accidents with core melt which would lead to early or large releases have to be practically eliminated; o for accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures. The limited protective measures that may be needed should be as follow: Measures Evacuation zone Sheltering zone Beyond sheltering zone Permanent relocation No No No Evacuation May be needed No No	Canada to meet the qualitatively safety objectives listed. c) The requirements for protection against commercial aircraft crash established in RD-337 (and the updated regulatory document REGDOC 2.5.2) provide a design that should not lead to core melt. The following acceptance criteria have been developed: The design shall provide for the ongoing availability of fundamental safety functions during BDBTs; these provisions will depend on the severity of the threat. For more severe events, there shall be a safe shutdown path that comprises at least one means for each of the following: 1. reactor shutdown 2. fuel cooling 3. retention of radioactivity from the reactor There shall be sufficient structural integrity to protect important systems. Two such success paths shall be identified where practical. For extreme events, there shall be at least one means of reactor shutdown and core cooling. Degradation of the containment barrier may allow the release of radioactive material; however, the degradation shall be limited. In these cases, the response shall include onsite

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				Sheltering May be needed May be needed No Iodine Prophylaxis May be needed May be needed No c) Intentional crash of commercial aircraft a crash should not lead to core melt and therefore not cause more than a minor radiological impact.	and offsite emergency measures.
159	Spain	Article 18.1	Pg. 182	Could you provide the percentage of CNSC resources that have been devoted to new reactors and to pre-project design review during last year? How is the CNSC pre-project work funded? Has CNSC capability of getting funds for this activity from fees paid directly by the vendors or is it necessary to go through the licensee?	Approximately 5% of technical resources have been allocated to pre-project vendor design reviews in the 2012-2013 fiscal year. Vendors pay for this service directly, under the terms of a service agreement. The participation of a potential applicant is not mandatory. A potential applicant may have their own arrangements with the vendor. The vendor resign review process does not fetter the Commission's discretion in regulatory matters.
160	Switzerland	Article 18.1	18 (i) / p.184	According to the statements given regarding compliance with article 18, the Canadian NPPs are well prepared with features that prevent accidents and help mitigate impacts in case of accidents. Such features include e.g, large inventories of cool water, two groups of independent, physically	All Canadian NPPs have separate seismically qualified emergency power and water systems. These systems are totally independent of regular and backup power and water systems that are used in regular plant operations and in response to anticipated operational occurrences. These systems also enter the plants and are field run separately from the

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				separated, and diverse backup power and cooling water systems. Switzerland is interested to learn how the diversity in power and cooling water supply is achieved.	regular and backup systems.
161	Switzerland	Article 18.1	p.319	Regarding the assessment of defence in depth in response to Fukushima in annex 18(i), CNSC is stating that in the event the steam generators are unavailable, the large inventory of cool water (moderator and the calandria vault/shield tank) that surrounds the fuel can provide passive cooling to prevent accident progression and provide adequate time for long-term mitigation of accidents. In this context, could you please provide some information on the time available until a severe core damage does occur?	Approximate timings for core damage are provided in appendix B of the CNSC Fukushima Task Force Report (http://www.nuclearsafety.gc.ca/pubs_catalogue/uploads/October-2011-CNSC-Fukushima-Task-Force-Report_e.pdf). The time of severe core damage varies with the station and with the success or failure of mitigating measures. The timing below is typical. Following a loss of all heat sinks, there is sufficient water in the boilers to provide cooling for about 1.5 hours. If the operator can initiate crash cooling of the boilers, this allows gravity feed which can extend boiler dryout for a further 4 hours. If boiler makeup is not recovered in this time, primary coolant will boil off leading to fuel overheating in about 4 hour. If primary coolant makeup can be established, core damage can be prevented. Cooling by the moderator can prevent severe core damage for a further 5.5 hours. If moderator makeup can be established, severe

Ser	Country	Original Reference	Reference in Report	Questions/Comment	Response
					core damage can be prevented. The molten core can still be retained in the calandria vessel with cooling from the shield tank water for several more hours, or indefinitely if makeup can be added.
162	United Kingdom	Article 18.1	Vendor pre-project design reviews	Please explain why CNSC only makes public a small amount of information, the executive summary, on the results of the vendor pre-project design review process.	The VDR process preserves vendor proprietary information while giving the public information through an Executive Summary. The review is solely intended to provide early feedback on the acceptability of selected aspects of a nuclear power plant design based on Canadian regulatory requirements and CNSC expectations. All key findings are provided in the executive summary. It is not a certification of a design and does not fetter the Commission in the licensing process. If the design were to be included in an application for a licence, the material provided for and developed in the review would be part of the record under the licensing process, and fully available to the public.
163	Indonesia	Article 18.2	p. 197/341 or p. 185	Measures are embedded in the Canadian licensing process to ensure the application of state-of the-art, proven technologies. In each phase of licensing, documents have to be submitted to describe the technology employed and to verify and validate it. These include the design and safety	The design authority shall identify the modern codes and standards that will be used for the plant design, and evaluate those codes and standards for applicability, adequacy, and sufficiency to the design of SSCs important to safety. Where needed, codes and standards shall be supplemented to ensure that the final quality of the design is commensurate with the necessary

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				analysis information in the safety analysis report and the QA program for design and safety analysis. How does the Government of Canada define "Proven Technologies" in the regulation or guides? Is it related to design certification process? And, is there any special process for the licensing of proven technology NPP?	safety functions. SSCs important to safety shall be of proven design, and shall be designed according to the standards and codes identified for the NPP. When a new SSC design, feature or engineering practice is introduced, adequate safety shall be demonstrated by a combination of supporting research and development programs and by examination of relevant experience from similar applications. An adequate qualification program shall be established to verify that the new design meets all applicable safety requirements. New designs shall be tested before being brought into service and shall be monitored while in service so as to verify that the expected behaviour is achieved. The design authority shall establish an adequate qualification program to verify that the new design meets all applicable safety design requirements. In the selection of equipment, due attention shall be given to spurious operation and to unsafe failure modes (e.g., failure to trip when necessary). Where the design has to accommodate an SSC failure, preference shall be given to equipment that exhibits known and predictable modes of failure, and that facilitates repair or replacement.

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164	France	Article 18.3	§ 18 - p.181 and 182	Could Canada specify how “Fukushima magnitudes” are chosen for external events? Is it by an increase of intensities of a given factor for the design intensity, or is it based on a large return period (which return period)?	For external hazards in most cases Licensees perform PRAs where the hazard is considered for a broad range of probability levels in order to be convolved with the plant fragility so that a risk estimate can be the resultant. In that sense there is not a single review level condition. In the case of seismic evaluations, for assessments where review level earthquake is applicable, the review level earthquake is to be one order of magnitude lower in terms of probability than the design basis level. It should be understood that review level condition is benchmarked against, but there is no requirement for SSCs to be able to withstand it. For information, for plant HCLPF from seismic beyond design basis, Canada is moving towards implementing a margin factor versus design basis earthquake (draft CSA N289.1 amendment).
ARTICLE 19: OPERATION					
165	Spain	Article 19.1	Pg. 191 and pg. 192	According to the report, all the NPP licensees are scheduled to complete the modification of critical limiting conditions by 2013.Could you confirm that all	1) The development of the SOE in all Canadian NPPs has been completed, and has now transitioned to a “Maintenance Mode”. The initial implementation drove the NPPs to an extensive review of their maintenance and

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				Canadians NPP licensees have completed the modification of critical limiting conditions and that an acceptable safe operating envelope (SOE) is available in all of them? Once the SOE is been established, have the licensees reviewed the maintenance and testing practices to make sure that all the relevant components of the safe operating envelop are able to fulfill its safety function? Has the regulatory body carried out any specific program to make sure that SOE are acceptable and are implemented in an appropriated manner?	testing practices. Compliance Tables were developed to ensure every identified item (in maintenance and testing) was tracked and driven to a satisfactory resolution. Thus, a strong SOE is considered to be in place. 2) Compliance tables were developed to ensure every identified item (e.g., surveillance and testing) is tracked as how it is implemented in operating documentation. The CSA N290.15 standard “Requirements for the Safe Operating Envelope of Nuclear Power Plants” was developed to document a standardized and consistent approach to defining, implementing and maintaining the SOE for an operating plant; this CSA standard is a means to bring consistency amongst licensees for these aspects of the SOE. 3) The CNSC carries out compliance activities to verify compliance over the licensing period. No significant findings have been identified.
166	Spain	Article 19.1	Pg. 196	Are currently available in the Canadian NPP guidelines and equipment to cope with big destructions and fires, as those that could be caused by the impact of a commercial aircraft?	CNSC regulatory documents, such as design safety analysis, accident management, emergency preparedness, site suitability are being updated to address commercial aircraft impacts. Licensees would use Emergency Mitigating Equipment (EME) and SAMG for these types of events. It is worth noting that for fires, all licensees have Fire Hazard Analysis and Fire Safe Shutdown Analysis as required by N293.

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167	Spain	Article 19.1	196	Are the knowledge and abilities to cope with a severe accident and the use of SAMG an item included in the CNSC oversight program and in the certification processes of the operating personnel? If this is not the case, do you plan to include it in the near future?	The CNSC expectations regarding certification of staff are outlined in Regulatory Document RD-204: Certification of Persons Working at Nuclear Power Plants. These requirements include the need for staff to be trained for operation and monitoring of NPP systems under normal, abnormal and emergency conditions. The full-scope simulators in use in Canadian NPPs cannot replicate all severe accident conditions, but all NPP licensees have severe accident management guidelines and certified staff receive initial, continuing and update training on their use. Licensees continue to explore activities to enhance this training. Existing regulatory guide G-306 establishes expectations that CNSC staff applies on our current evaluations of the SAMG. A revision of this document, to be called REGDOC-2.3.2, will provide a strengthened and expanded regulatory framework for the regulatory oversight of accident management, including requirements for the plant personnel. However, we do not anticipate any formal certification of the personnel with regards to SAMG.
168	Spain	Article 19.1	Pg. 196	Are the knowledge and abilities to cope with a severe accident and the use of SAMG an item included in the CNSC oversight program	Repeat of question #167.

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				and in the certification processes of the operating personnel? If this is not the case, do you plan to include it in the near future?	
169	China	Article 19.2	P190	It seems to me that the OP&P plus SOE equals PWR's Technical Specification. Could you please give a detailed explanation about the difference between PWR's Technical Specification, OP&P and SOE?	While this is a good analogy, the SOE is actually more in depth than Technical Specifications. Canadian practice of safe operating envelope (SOE) is equivalent to the approach of PWR's Technical Specification such as requirements for technical specifications as per USNRC 10CFR 50.36, and also equivalent to the approach of Operating Limits and Conditions (OLCs) as per IAEA SSR-2/1. Although the terminology of approaches varies, the intent and substance of the SOE have always been fulfilling regulatory requirements to make sure that the plant operates within SOE that is supported by the design and the safety assessment. As reported in the Canadian National Report, to ensure consistent practice between Canadian plants, the CNSC required every nuclear licencee in Canada to use the SOE approach as per Canadian CSA standard N290.15, "Requirements for the Safe Operating Envelope of Nuclear Power Plants". Subsequently the SOE has been implemented in all Canadian NPPs, and regular CNSC inspections are carried out to verify compliance.

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170	China	Article 19.2	P192	Page P192mentioned the each safety function is operating correctly and meets availability limits (typically, 99.9 percent), which means that their unavailability is less than 0.001. The four special safety systems' unavailability is also less than 0.001 regulated in R7, R8, R9, which is relatively strict for other safety systems. Could you please give some introduction about the actual observing unavailability value?	Observed Unavailability for Canadian NPPs are recorded based on station records and further analysis to determine when Special Safety Systems are impaired and considered not to meet their design intent. The unavailability values are typically much smaller than their targets but occasionally their values may exceed their targets.
171	Czech Republic	Article 19.2	Page 190	What is the connection (difference) between L&C (Limits and Conditions) and OP&P (Operating Policies and Principles)?	Most of our OP&Ps are based on good Engineering Judgment. The development of our SOE produced well documented Safety Limits. From our perspective, Safety Limits are one step deeper in knowledge than L&Cs. Canadian practice follows the safe operating envelope (SOE) approach as per Canadian CSA standard N290.15-10, "*Requirements for the Safe Operating Envelope of Nuclear Power Plants*". Subsequently the SOE has been implemented in all Canadian NPPs. Canadian practice of using the SOE approach includes OP&P, Operating Manuals and Operational Safety Requirements to provide required scope and details of the SOE. The Operating Limits and Conditions (OLCs) are specified in OP&P at a high level and Operating Manuals and

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					Operational Safety Requirements at a detail level for all safety systems and safety-related systems.
172	Japan	Article 19.4	19(iv), p195	Canadian report says, "the CNSC identified a need to more explicitly define the regulatory requirements for accident management." How do you define and require activities to be carried out during severe accident to maintain the integrity of components which might be indispensable for severe accident management?	Regulatory document REGDOC-2.3.2 is in its final stages of development and is planned for issue in 2014. This REGDOC will establish regulatory requirements for an integrated accident management programme, which addresses all types of accidents, considers diverse initiating events (internal and external), and identifies various objectives, including protection of physical barriers to prevent release of radioactivity, as well as the need for design provisions to achieve accident management objectives.
173	Korea, Republic of	Article 19.6	198	Please explain in detail the guidance for "screening of issues that are to be shared with the public".	Essentially, issues are screened to identify the appropriate communication vehicle to be used. The CNSC develops, selects and uses guidance depending on the specific situation being screened. Our Emergency Response Manual provides guidance on major events and includes a list of public organizations to be contacted. The CNSC Event Initial Reporting Process document guides staff on when to submit event details to our publicly held Commission meetings. The process guidance is provided with specific examples including, but not limited to, exposure to individuals, death or injury, unplanned release, actuation of safety

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					systems and others. Annually, the CNSC publishes an Integrated Safety Assessment of Canadian Nuclear Power Plants that includes specific or summary information of events reported by licensees.
174	France	Article 19.7	§ 19.7 - p.199	Could Canada mention the criteria defining significant events compared to minor events? Could Canada give examples of corrective actions taken further to international operation experience during the reporting period?	The CNSC develops, selects and uses guidance depending on the specific situation being screened. Our Emergency Response Manual provides guidance on major events and includes a list of public organizations to be contacted. The CNSC Event Initial Reporting Process document guides staff on when to submit event details to our publicly held Commission meetings. The process guidance is provided with specific examples including, but not limited to, exposure to individuals, death or injury, unplanned release, actuation of safety systems and others. Annually the CNSC publishes an Integrated Safety Assessment Report on Canadian Power Plants that includes specific or summary information of events reported by licensees. Canadian Power Reactor Licences include regulatory reporting requirements under S-99. "Reporting Requirements for Operating Nuclear Power Plants. Licensee reports are reviewed in the context of the compliance baseline activities so that reactive inspections can be initiated as appropriate to review or

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					measure compliance with regard to a specific event or situation. This regulatory vehicle is also used to follow-up on corrective actions
175	Russian Federation	Article 19.7	p.199	The Report presents information on the process of operating experience feedback. Do the Operator and Regulator use any criteria/ indicators to evaluate the effectiveness of this activity (operating experience feedback)?	A series of inspection checklists have been developed by CNSC specialists to aid them in confirming that licensee operating experience programs meet the requirements stated in CSA standard N286.5 'Operations Quality Assurance for Nuclear Power Plants' and S-99 'Reporting Requirements for Operating Nuclear Power Plants'. These checklists include criteria for evaluating event investigation processes, the effectiveness of corrective actions, and the trending and analysis of events.
176	Korea, Republic of	Article 19.8	200	On page 200 in the Article 19, it is stated that "The Canadian nuclear industry minimizes waste through enhanced waste monitoring capabilities to reduce inclusion of non-radioactive wastes in radioactive waste". What measures do you use to prevent release of radioactive waste mixed with non-radioactive waste to the environment under uncontrolled circumstances?	All material that is removed from Canadian NPPs is monitored for radiation before being released into the public domain. The Canadian NPPs use a defence in depth philosophy to ensure that radioactive waste is not unintentionally removed from the NPP with non-radioactive waste. For the waste streams, very large items are monitored by hand and verified by a second qualified person. Other large and smaller objects are monitored by hand or through monitors (including drums). All waste bags are also put through monitors. In addition to these verifications, all vehicles transporting non-radioactive waste travel through vehicle monitors before leaving the

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					NPP site. For example, OPG’s process determines active from inactive wastes and then processes that material for shipment off site. The basis limits are established by the CNSC and monitored by OPG’s Health Physics department. Material is segregated and controlled in separate areas to prevent any cross contamination. Under “uncontrolled circumstances” there would be no change to OPG processes except for enhanced monitoring of inactive waste streams, assuming OPG had some concern that contaminated material could potentially be released externally. In a severe scenario, shipping of material would be strictly minimized to ensure control of material.