CONSEJO DE SEGURIDAD NUCLEAR

Preguntas formuladas por España a las Partes Contratantes en la séptima reunión de revisión de la Convención sobre Seguridad Nuclear

Convention on Nuclear Safety

Questions Posted By Spain in 2017

No.	Country	Article	Ref. in National Report	Question	Answer	Support Documents
32	Belgium	General	page 9	PSA development • When is expected to complete the development of the Fire & Flooding L2 PSA for all the Belgian units? • Will they be plant specifc PSAs or adaptations of the one mentioned in the report?	The deadline for the Fire and Flooding PSA-level 2 of the NPPs was 01/01/2016 – this requirement was defined in the framework of the WENRA RL 2008. The studies and the results were introduced by the licensee on time, for all units, except for Doel 1/2. After analysis, the studies and the results of the flooding level 2 PSAs, were considered acceptable, including the fact that they have been performed for representative units. For the fire PSAs, the PSAs were considered too conservative to really reflect the real risk of the units. New deadlines were imposed to the licensee to update the Fire PSAs by end 2017. For Doel 1/2, as it was initially foreseen to definitively close these units in 2015, the studies were not performed by the end of 2015: the best estimate planning is mid-2017.	
33	Belgium	General	page 9	PSA development • Which is the update frequency of the Belgian PSA´s?	Belgian PSAs are updated every 5 years. More precise, an “update” is made every 5 years taking into account modifications to the installations and experience feedback for the data. Then a major upgrade occurs after 10 years. For this upgrade, the PSA-models/methodologies are also improved	

34	Belgium	General	page 10	Periodic Safety review INSAG NS-G-2.10 has been used to perform the PSR of some of the Belgian plants. According to the methodology described in the mentioned IAEA document, standards and good practices must be identified in order to assess every Safety Factor against them. Please, elaborate: • What criteria were used to select these standards and good practices? • Were the type and sources of these standards established a priori o defined specifically for each Safety Factor? • Could you provide some examples for some Safety Factors?	The selected standards for further consideration in the PSR are those related to the regulations and guides taken into account in Belgium according to the applicability status thereof. A list of 'Good Practices' is established, based upon those that can be found in the databases of the following three institutions: -World Association of Nuclear Operators (WANO); -Institute for Nuclear Power Operations (INPO, US); -OSART Mission Results (OSMIR) Publications which are not relevant with regard to design, operation and management of the unit as well as publications on domains which are out of scope of the PSR framework (e.g. security, safeguard) are discarded. Relevant regulations and standards with regard to design, operation and management were analyzed prior to the PSR. One or several Safety Factors are associated with publications considered to build the update of the regulation framework and to Good Practices as well. Some examples of selected good practices: -INPO OR.4 "Management and leadershipdevelopment"(SF10 and SF12 assessments); -WANO GP ATL 02-001 Control of lifting, rigging and cranes (SF 10,3,2).	

25	Brazil	Article 6	page 40	In the above mentioned page is said that one modification (from the ETN Fukushima Response Plan) has been the interconnection of the bus bars of the Emergency Power Supply D2 (power supply by small Diesel Generator set) with the bus bars of the Emergency Power Supply D1 (power supply by the large Diesel Generator set What is the purpose (functionality) of this interconnection? Could you please provide more specific information about the design of this interconnection and how it may change or not the original design functionality?.	The purpose of the interconnection of D1 x D2 was to increase the availability of the Emergency Diesel Power Supply Systems (EPSS1 – Emergency Power Supply System 1, 4x6.600 KVA DG(D1) and EPSS2- Emergency Power Supply System 2, 4x1050 KVA DGs(D2)) of the Angra 2 plant, in operation and emergency power cases. In the original design in case of loss of offsite power both Diesel generator sets would start to supply the required loads. With this configuration, some loads were supplied only by the EPSS2. PSA studies indicated that loss of one of the EPSS2 DG had a large impact on the Plant Core Damage Frequency, basically because of the importance of the equipment supplied only by the EPSS2 for the control of the different accident sequences. With the implementation this interconnection, in case of Loss of Offsite Power, only the Emergency Power Supply Diesel Systems D1 starts, supplying power to its own loads as well as to the Emergency Power Supply System D2 busbars, providing double power supply to the EPSS2 loads. The Emergency Diesel Load Programs D1 follow their designed starting time parameters: 2s waiting time (U<0,8Un or f<56,7 Hz on a 60 Hz base system) before DG start , 10 seconds DG starting time until rated speed is reached and the busbar being fed reaches operating voltage. Thereafter the DGs 1 load program time starts running (after 12s of onset of loss of voltage or	

26	Brazil	Article 6	page 46	In the above mentioned page is said that one additional diesel generating set, similar to the existing ones in the emergency diesel building, shall be included in the plant design for Angra 3 Do this additional diesel generator already exist in the actual plant design of Angra 1 and Angra 2.? If not, has it been considered or assessed the implementation of this modification also in these plants?	No, this additional diesel generator (DG) does not exist in the actual design of Angra 1 and 2. Concerning the second question, below is explained why the implementation of such modification was not considered. As additional information, relevant for the explanation below, the Brazilian Regulator adopts the rules and regulations of the Country supplying the NPP when no applicable national rule is available. Angra 1 (US, Westinghouse design): In addition to the two original DGs, two additional ones of even larger size, meeting all the requirements for an emergency Diesel, have been incorporated to the Plant in its early stage of operation. That means that Angra 1 has today more than 4x100% DG redundancy. Accordingly, besides following Brazilian and USNRC regulations, installation of another DG is not justified. Angra 2 (Germany, KWU design): Having the same basic design of Angra 3, this plant have the same DG configuration, that is 4 large DGs, that provide 4x50% capability(EPSS1) for all emergency conditions plus 4 smaller DG (EPSS2- black out DGs), also 4x50% capability, for plant cooling in case of LOOP associated with loss of the large DGs. To take into account the KTA requirement of an additional power supply source after 72 hours, when it is assumed that the existing DGs from EPSS 1 and 2 become unreliable, Angra 2 has available two mobile	
57	China	Article 6	page 26	In the report (page 26) is mentioned the existence of containment filtered venting for HPR1000 design plants. It does not mention if that feature already exists or not in other design existing NPP. Has it been assessed, and what were the conclusions, the convenience of requiring the implementation of containment filtered venting in those plants that don´t have it, as a post-Fukushima action?	All the operating nuclear power plants in China have been assessed. The necessity to add containment filtration and discharge system was determined based on assessment results by considering the possibility of containment over-pressure accident. In general, the filtration and discharge system is not required if the measures have been taken to prevent the reaction of core melting with lower chamber concrete (MCCI) that result in containment over-pressure.	

32	Finland	Article 6	page 19	Regarding the extension of the original design lifetime for Olkiluoto NPP that was 40 years, which is the new lifetime period considered for both units?	TVO (the licencee of the Olkiluoto 1&2) left in 26.1.2017 the application of the renewal of the operating licence to the Finnish Ministry of Economic Affairs and Employment concerning the 20 years lifetime extention. Current operating licence is valid to the end of 2018.	
33	Finland	Article 6	page 18	Regarding the Loviisa reactor pressure vessels, which modifications have been made at both units to reduce the brittle fracture risk?	Reannealing has been done for Loviisa 1 in 1996, but not for Loviisa 2. Margins has been analysed (with the deterministic and propabilistic embrittlement analyses) and LTO was approved in 2007. In the recent deterministic analyses (used in PSR 2015) the deterministic embrittlement temperature margin was decrased some degrees because of the changes in Loviisa I&C renewal project (affecting to assumption of the possible loads). The embrittlement temperature margins were enough for the Loviisa 1 but for Loviisa 2 very close to the aproval limit. STUK required as a part of the PRS inspection the licencee to send at the end of the 2016 the report how to increase the embrittlement margins at Loviisa 2. The low margins at the Loviisa 2 are especially involved to the event where RPV's core area weld seam outer surface is cooling while unexpected start of the sprinkler system of the reactor building occurs. Concerning the licencees report the one corrective action is to modify the sprinkler system's cooling unit function to increase the initial temperature of the sprinkled water (planned to implement in 2019). The licensee continues also the investigation of the opportunities to isolate the RPV's core area weld seam outer surface. Licensee will update the propabilistic and the deterministic embrittlement analyses before the next PSR 2023 so the influence of the corrective actions can be identified then.	

53	France	Article 6	page 39	• Which are the most important lines of work for addressing the obsolescence of the I&C hardware through the renovation of certain equipment which would be unable to reach a 40-year service life? • Is it planned to participate in international existing programs regarding this issue or promoting new ones?	Obsolosence and ageing are important issues, the Periodic Safety Review (PSR) is a particular opportunity for an in-depth examination (see 14.2.1.4), especially starting from the third PSR for French NPPs. Very few equipments would be unable to reach a 40-year service life. The issue is more for long-term service life, beyond 40 years. For I&C harware which would be unable to reach a 40 year service life, the main topics and the strategy are the following : - ageing of connections (survey, tests of samples…); - ability to provide for additionnal capacity, i.e. capability of I&C systems to embed new functions : is it possible to add new Input/ouput, to perform new functions (CPU load) ? It could be a raison to retrofit; - availability of spare parts : relationship with our suppliers to get spares part (last buy order) for repair and replacement of hardware (EDF tries to implement long term maintenance contracts for I&C hardware); - efforts to redesign using the installed technology in order to avoid important retrofit. EDF/R&D works with EPRI (USA), participates to IEC committees, EXERA commission , AFCEN and to a working group involving the main French industrials companies facing the same technical issue (I&C hardware ageing) : Department of Defense, Airbus,	

34	Germany	Article 6	page 40	In this page is said that besides fundamental provisions regarding the scope and depth of the analysis methods, the requirements listed in the guideline (for the performance of integrated event analysis) the requirements listed in the guideline also comprise organizational requirements for the license holders of the nuclear installations. Could you provide some more specific information about what these organizational requirements demand	The guideline for the performance of integrated event analyses has the following organisational requirements: - The event analysis has to be integrated in the safety management system - The licensee has to define unambiguous requirements how the event analysis is to be performed and how the results are to be used. This has to be communicated as part of the code of conduct to all employees. - An appropriate team of expert has to formed that is reinforced by experienced employees of different departments on a case-by-case basis. - The general management has to equip the event analysis team with the necessary authority for performing the event analysis. Further, the guideline makes demands on human resources, tools and infrastructure, on the organisational/structural integration of the analysis team (for example, it has to be ensured that the analysis team has access to all information and personnel, irrespective of organisational hierarchy) and on the timetable of the analysis.	

6	Portugal	Article 6	research reactor	Could you please explain your forecasts regarding the operation and utilization of the research reactor in the medium and long term? What human and financial resources you have to support the future operation of the installation, taking into account the implementation of the INSARR mission recommendations?	During the last ten years, the reactor has operated at full power (1 MW) one week per month, on the average. Therefore, the current fuel may steel be used for another ten years of operation. The human and financial resources for the implementation of the INSARR recommendations are provided by IST and by the Portuguese Government, as in article 4(3) of Decree-Law 29/2012 which foresees Government support in the case of refurbishment and decommissioning of the RPI. Nevertheless, IST has to submit to the Government a plan for the future operation of the RPI that covers research, education and training, and services. This plan was suggested after a scientific peer review mission that took place in December 2015 at the request of the Portuguese Foundation for Science and Technology. This plan for the future operation of the RPI, together with the implementation of the INSARR recommendations is the challenge that IST faces now.	
27	Sweden	Article 6	page 16	It is stated that in the year 2015 was decided the phase-out of the reactors Ringhals units 1/2 and Oskarshamn units 1/2. The decision was taken in respect, among others, of SSM´s safety requirements regarding operation beyond 2020. Could you please provide information on the origin of these safety requirements (Long Term Operation regulations, specific safety regulations…)?	The new requirements regarding installation of full independent core-cooling system was motivated by the accident at Forsmark NPP in 2006, but was raised again in connection to EU stress test. The dependency on supply of electric power in case of an emergency at the Swedish reactor units has been discussed already in 90´s. An extra and fully independent system was subject of discussions already at that time. The results of the stress tests and subsequent analyses and conclusions resulted in the regulatory decision to install such systems, which are required to be in place for the continued operation of units after 2020.	

28	Switzerland	Article 6	page 13	In late 2013, it was announced that Mühleberg NPP will be decommissioned at the end of 2019. ENSI has developed the guideline G17 "Decommissioning of nuclear facilities". Could you please explain whether the above mentioned guide considers aspects related with transition of operating reactors plants to decommissioning? If not, are there standards or provisions for developing guidance to facilitate transition?	The guideline ENSI-G17 defines the requirements for the decommissioning in several phases including the transition phase.	
71	United States of America	Article 6	67, paragraph 4	Regarding the Reactor Oversight Process annual self-assessment, it is mentioned that it was redesigned in 2015 to develop a more effective process. Why do you think it was not being as effective as it could be and which are the "specific areas of interest" that were reviewed in order to improve the process?	As noted in SECY-14-0047, "Reactor Oversight Process Self-Assessment for Calendar Year (CY) 2013," dated April 18, 2014 (ADAMS Accession No. ML14066A365), the NRC staff had initiated its ROP enhancement efforts to take a "fresh look" at several key areas of the ROP, including but not limited to the self-assessment program. In addition, in CY 2013, the ROP benefited from independent evaluations by the Government Accountability Office, the Office of the Inspector General, and a Commission-directed internal independent review. These efforts collectively produced numerous recommendations and suggestions for further ROP improvements, including improvements to the self-assessment process itself. For example, a specific recommendation from the Commission-directed independent review, "Reactor Oversight Process Independent Assessment 2013" (ADAMS Accession No. ML14035A571), was to revise the ROP self-assessment process to better solicit and assess both tactical and strategic feedback. Given the amount of feedback and recommendations received by independent evaluations, staff recognized that the prior self-assessment process did not provide as deep of a review as necessary to identify some of these underlying enhancement opportunities. In 2015, the NRC staff completed the redesign of the	

72	United States of America	Article 6	153	Audits and vendors supplies How do you verify the effectiveness of the supply chains? Have you implemented tools to address counterfeit and fraudulent items in nuclear facilities? Just in case, please describe them.	As required by 10 CFR Part 50, Appendix B, U.S. nuclear reactor facilities are responsible for the establishment and execution of a quality assurance program. They may delegate activities to others (e.g., contractors, agents, and consultants), but they retain the responsibility for quality assurance. U.S. nuclear reactor facilities are also required to control purchased material, equipment, and services through audits, surveys, and inspections at routine intervals based on importance, complexity, and quantity of products or services. The NRC also conducts vendor inspections at companies that supply materials, equipment, and services under a 10 CFR Part 50, Appendix B, quality assurance program. The results of these inspections are communicated to the vendor and the U.S. nuclear reactor facilities to highlight weaknesses in the nuclear supply chain and supply chain oversight. NRC vendor inspection reports are publicly available at: https://www.nrc.gov/reactors/new-reactors/oversight/quality-assurance/vendor-insp/insp-reports.html. Although supply chains for other industrial sectors may be substantially affected by Counterfeit, Fraudulent, and Suspect Items (CFSI) events, it is the NRC’s position that adherence to existing NRC regulations provides adequate protection of the public health and safety. Specifically, if a U.S. nuclear reactor facility implements	

73	United States of America	Article 6	page 67	Regarding the Reactor Oversight Process annual self-assessment, it is mentioned that it was redesigned in 2015 to develop a more effective process. Why do you think it was not being as effective as it could be and which are the “specific areas of interest” that were reviewed in order to improve the process? Regarding the Reactor Oversight Process annual self-assessment, it is mentioned that it was redesigned in 2015 to develop a more effective process. Why do you think it was not being as effective as it could be and which are the “specific areas of interest” that were reviewed in order to improve the process?	As noted in SECY-14-0047, “Reactor Oversight Process Self-Assessment for Calendar Year (CY) 2013,” dated April 18, 2014 (ADAMS Accession No. ML14066A365), the NRC staff had initiated its ROP enhancement efforts to take a “fresh look” at several key areas of the ROP, including but not limited to the self-assessment program. In addition, in CY 2013, the ROP benefited from independent evaluations by the Government Accountability Office, the Office of the Inspector General, and a Commission-directed internal independent review. These efforts collectively produced numerous recommendations and suggestions for further ROP improvements, including improvements to the self-assessment process itself. For example, a specific recommendation from the Commission-directed independent review, “Reactor Oversight Process Independent Assessment 2013” (ADAMS Accession No. ML14035A571), was to revise the ROP self-assessment process to better solicit and assess both tactical and strategic feedback. Given the amount of feedback and recommendations received by independent evaluations, staff recognized that the prior self-assessment process did not provide as deep of a review as necessary to identify some of these underlying enhancement opportunities. In 2015, the NRC staff completed the redesign of the	

74	United States of America	Article 6	page 78	Please, could you provide additional information on this statement under Vienna declaration on nuclear safety?: Because NRC requirements protect public health and safety through prevention of accidents and by mitigating releases in the event of an accident, the risk of offsite contamination is rendered acceptably low as an indirect benefit, rather than as a direct performance goal	The NRC regulatory framework requires that plants be designed with multiple independent and redundant safety systems. Plants must also be designed with multiple barriers including a reactor containment to prevent a radioactive release and be designed with systems that would mitigate any potential releases. These features provide a “defense-in-depth” approach that reduces the probability of reactor accidents and precludes a large release. To further minimize the risk of an accident, nuclear power plant operators are required to be highly trained and skilled personnel that undergo continual training and testing. This layered approach has been successful in ensuring that plants are designed and operated safely in the U.S. While there have been a small number of incidents at nuclear facilities, because of these regulatory requirements none of them have resulted in a large release to the public or the environment. In addition to the safety features of a nuclear power plant, the NRC requires licensees to establish emergency preparedness plans to assure that protective measures can be taken to protect the public in the unlikely event of a radiological emergency. In the unlikely event of an emergency these plans will guide the response including assessing the consequences of the event, promptly notifying the public, and determining protective	
40	Finland	Article 7	page 25	In 2014 an assessment of the applicability of new safety guides to the operating plants was done. • Do the guides contain guidance for this exercise? • How is it decided when an improvement to an operating unit is reasonably practicable?	There is no guidance to the exercise for comparison. The judgement of reasonable practicable improvements is based on licensees' evaluations and regulatory review of the possible improvements. Among other things, the safety significance, and the complexity of the improvement and the possible drawbacks of the implementation are taken into account when making the judgement.	

41	Finland	Article 7	page 24	The report states: The regulatory guides are continuously re-evaluated for updating. If there is not any immediate need for corrections or updates of YVL guides (e.g. new international requirements or update of pertinent national legislation) there are criteria for the review and updating of the regulations Could you, please, provide additional information on the stablished criteria for the review and updating of the YVL guides, or regualtions in general	Some needs for improvement come from the updated safety reference levels (including WENRA RLs and IAEA safety standards). Some of the requirements were seen not so well formulated during the enforcement how the new requirements should be implemented in existing plants. European directives may have some effects, as well as taking into account some changes in other areas of Finnish legislation. Most of the current needs are due to clarification of the requirements. There are, of course, needs for improvement in future, as well, but these are not urgent changes. The update needs come from experience in regulatory activities, from international requirements and from feedback from the licensees and other interested parties. It is said in the internal STUK instructions that the need for update shall be checked regularly.	
2	Senegal	Article 7	page 5, 14	According to paragraph “Anexes”, some information about laws and regulations is attached to the national report. Could you please check whether the information submitted is correct?		
8	Portugal	Article 7.1	page 9, 4ª paragraph	Once the Regulatory Commission for the Safety of Nuclear Installations (COMRSIN) was created as an independent regulatory body by Decree-Law 30/2012, have you planned to request an IRRS mission to assess the Portuguese regulatory system?	COMRSIN has prepared a letter to the IAEA, dated January 31st,requesting an IRRS mission for Portugal. This letter waits approval from the Minister of Science Technology and Higher Education because such review mission involves different agencies from different Ministries.	

78	Russian Federation	Article 8	article 8.2	Could you please explain which are the main steps of the decision-making process within the Regulatory Body? Within this decision-making process, how are managed technical discrepancies? Could you please explain how is the recruitment process in the Regulatory Body (Rostechnadzor)? (i.e. type of competition, weighting of experience, education, specific competitive exam, etc.)	The competition is conducted in two stages. The initial stage involves testing of the applicant compliance with the qualification requirements (the level of professional training, the length of state civil service (public service of other types) or the record (experience) of service in a job, expertise and professional skills required for the performance of respective duties; At the second stage the competition committee: a) assesses the applicants based on the documents they have provided concerning their education and civil service (other public service type) or any other labor experience, and decides if they meet the requirements existing for the civil service position the applicants apply for; b) assesses the professional and personal qualities of the applicants based on the selected competition procedures. The competition is conducted: a) in the form of individual interviews based on questions relating to the performance of the respective civil service duties; b) in the form of the applicant testing based on a single list of theoretical questions relating to the performance of duties for the civil service position. Based on the competition results, an order is issued by the employer's representative as to the appointment of the competition winner for the civil service position and	
42	Brazil	Article 9	page 79	Your report refers to licensing process as a mechanism to ensure that the regulatory requirements are fulfilled by licensees. Do NPP operating licenses include specific provisions or requirements related to the ways to be used by the licensee to fulfill its prime responsibility regarding safety?.	The operator's prime responsibility regarding safety is addressed in the CNEN's Safety Policy. The License issued usually state that: 1 - "the operator has to operate the installation according to the CNEN's requirements and the Technical Specifications that cannot be changed without CNEN's approval" 2 – "the operator has to have an organizational structure able to operate safely the plant, to comply with the requirements and to maintain Safety Culture patterns.	

97	China	Article 9	page 64-65	Please provide some information on how Civil Liability for Nuclear Damage is applied and the position of your country in relation to the Vienna convention. Are there national regulations on this matter?.	China is actively studying the accession to the relevant international conventions. Regarding the national regulations on this matter, the state council released An Official Reply on Nuclear Accident Damage Compensation Liability. According to the regulations, nuclear power plant operators shall assume absolute responsibility for nuclear damage accidents and the maximum compensation for nuclear damage resulted from one nuclear accident is RMB 300 million Yuan. If the total payable compensation exceeds the maximum amount, the maximum state fiscal compensation is 800 million Yuan. It will be discussed case by case for the damage needed greater compensation liability.	

61	Finland	Article 9	page 35-36	Please provide some information on whether the licensing process and the terms and conditions of the license are used in Finland as a way to ensure that the license holder complies with its obligations regarding safety.	Nuclear Energy Act Section 7 f states that construction and operation safety shall take priority during the construction and operation of a nuclear facility. The holder of a construction licence shall be responsible for the nuclear facility's construction in accordance with safety requirements.The holder of an operating licence shall be responsible for the nuclear facility's operation in accordance with safety requirements. The requirements for the license application files submitted to STUK for the safety review are given in Nuclear Energy Degree (section 35 for the construction license and 36 for the operating license). Prior submitting the files to STUK the conformance and acceptability of the documents pertaining to safety-significant products submitted to STUK shall first be duly reviewed by the licensee’s in-house organisation. The same principle is followed during the whole licensing process of structures, systems and component - license applicant's / licensee's own safety assessment is mandatory part of documentation when approvals from STUK are asked. Principles for the safety assessment required are given in the YVL guides B.1 (safety assessment independent of the designer drawn up by the licensee) and A.1 (summary of justifications).	
97	France	Article 9	page79-80	Please provide some information on how Civil Liability for Nuclear Damage is applied and the position of your country in relation to the Vienna convention. Are there national regulations on this matter?.	The provisions applicable to civil liability in the field of nuclear energy are the subject of a special chapter in the Environment Code (Article L. 597-1 to Article L. 597-46). A common protocol for the application of the Vienna Convention and the Paris Convention was adopted in 1988. It makes it possible to extend the compensation regime of a Convention to the victims of the Contracting Parties to the other Convention. This protocol, ratified by France, entered into force on 30 July 2014.	

59	Germany	Article 9	page 78-81	Please provide some information on how Civil Liability for Nuclear Damage is applied and the position of your country in relation to the Vienna convention. Are there national regulations on this matter?	Germany is contracting party to the 1960 Paris Convention on Third Party Liability in the Field of Nuclear Energy (Paris Convention). The Paris Convention is directly applicable in Germany. It establishes a comprehensive regime for civil liability for nuclear damage. Under the Paris Convention the nuclear installation operator is exclusively liable for nuclear damage that is caused by a nuclear incident at his installation. Furthermore, the liability is strict, i.e. the nuclear installation operator is liable regardless of whether fault can be established. In addition to the provisions of the Paris Convention, Articles 25 to 40 of the Atomic Energy Act apply to the liability of the operator of a nuclear installation under the Paris Convention. According to Article 31 Paragraph (1) Atomic Energy Act, the liability of the operator of a nuclear installation under the Paris Convention shall be unlimited. In addition, Germany is contracting party to the 1988 Joint Protocol Relating to the Application of the Vienna Convention and the Paris Convention (Joint Protocol). The Joint Protocol links the 1963 Vienna Convention on Civil Liability for Nuclear Damage to the Paris Convention for the purpose of ensuring that the benefits of one Convention are also extended to the Parties to the other Convention.	

89	Russian Federation	Article 9	page 50	Could you please provide further details on how is sized (i.e. amount based on coverage) the financial coverage that is submitted to Rostechnadzor before obtaining an operating license?	With respect to the financial coverage for nuclear liability, the operation company (OC) is governed by Federal Law No. 170-FZ and the 1963 Vienna Convention on Civil Liability for Nuclear Damage. The financial coverage size is defined by the OC in accordance with the minimum limit of the nuclear plant operator's liability established by the Vienna Convention and amounting to 5 million USD as of 29 April 1963. The minimum limit is calculated annually with regard for the price of gold at the Central Bank's exchange rate. In accordance with Section 56 of Federal Law No. 170-FZ, the OC's financial coverage is formed by a government or another guarantee, the organization's own funds and an insurance policy (contract). The OC uses two types of financial coverage for the civil liability to third persons with respect to the damage and loss inflicted by a radiological impact, information on which is delivered to Rostechnadzor as the documented confirmation for the financial coverage: - a contract of insurance of the OC's civil liability for nuclear damage with the amount of coverage equal to the minimum limit of liability as defined by the Vienna Convention, - the organization's own funds in the amount of not less than the minimum limit of liability as defined by the Vienna Convention.	
58	Sweden	Article 9	page 85-88	Please provide some information on the mechanisms by means of which the regulatory body ensures that the license holder complies with its obligations regarding safety.	In principle, this question is about all activities carried out by SSM. Supervision is performed by inspections, safety reviews and in some areas supported by research. SSM follows operational events and any deviation observed in the licensees' organisations. A yearly report is written for each licensee and on a ten-year basis the periodic safety reviews summarises the situation at each plant.	

48	Switzerland	Article 9	page 53-54	Please provide some information on how Civil Liability for Nuclear Damage is applied and the position of your country in relation to the Vienna convention. Are there national regulations on this matter?	Switzerland has not signed the Vienna convention. The liability on nuclear accidents is governed by the national Nuclear Energy Third Party Liability Act and the corresponding Ordinance dated 18 March 1983 and 5 December 1983 respectively. According to these the operator is liable for any nuclear accident that occurs in the NPP without limitation (principles of strict liability, unlimited liability, channelling of the liability to the operator of a nuclear installation). The owner of a nuclear installation located in Switzerland is liable for nuclear damage abroad up to the amount that the national legislation of the state concerned provides for in relation to Switzerland (principle of reciprocity). The operator is obliged to insure nuclear accidents in the amount of CHF 1 billion. On 13 June 2008, Switzerland approved the revised Nuclear Energy Third Party Liability Act, subsequently ratifying the international Paris and Brussels Conventions. The revision of the Act increases the level of compulsory insurance coverage for nuclear accidents from CHF 1 billion to € 1.2 billion. It also greatly simplifies the claims procedure and so better protects victims' interests. On 25 March 2015, the Federal Council approved the revised Nuclear Energy Third Party Liability Ordinance. The Ordinance sets the minimum amount to be covered by private insurers at CHF 1 billion and specifies the risks which insurers are permitted to exclude. It also	

24	United Arab Emirates	Article 9	page 39-42	Please provide information on the situation of your country related to the Vienna Convention on Civil Liability for Nuclear Damage. Are there any plans to develop national regulations on this matter?.	The Federal Law by Decree No. 4 of 2012, “Concerning Civil Liability for Nuclear Damage” was established to determine civil liability and compensation for nuclear damage in the UAE. This Law in effect adopts the obligations and principles contained in the Vienna Convention on Civil Liability for Nuclear Damage as amended by the 1997 Protocol, which was ratified by the UAE. Federal Law by Decree No. 4 of 2012 stipulates the legal requirements for operators of nuclear installation with regard to civil liability for nuclear damage. Under the Federal Law by Decree No. 4 of 2012, the operator of a nuclear installation is solely liable for any nuclear damage caused by a nuclear incident. In accordance with the provisions of the Federal Law by Decree No. 4 of 2012, the Federal Authority for Nuclear Regulation (FANR) is the competent authority in the UAE with respect to implementation of the provisions of this law, including, among others, issuance of rules and regulations relating to the application of the provisions of this Law. There are no current plans at FANR for issuance of regulations on this matter. As mandated by the Federal Law by Decree No. 4 of 2012, FANR has requested applicants for licences to	

105	United States of America	Article 9	113-114	A description of elements required to licensees to comply with their obligations is provided, including compliance with regulations and terms and conditions of the license, personnel training and qualification and openness and transparency. Is there any requirement in the US for the licensee to develop and maintain a management system, including the mentioned elements and others to comply with their obligations for safety?.	Each licensee is required to develop and implement a quality assurance program which complies with the requirements in Appendix B to 10 CFR Part 50 – "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." This program shall be documented by written policies, procedures, or instructions and shall be carried out throughout plant life in accordance with those policies, procedures, or instructions. The applicant shall identify the structures, systems, and components to be covered by the quality assurance program and the major organizations participating in the program, together with the designated functions of these organizations. The quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety. While the requirements of GS-R-3 cover management systems for regulatory bodies, the requirements of Appendix B to 10 CFR Part 50 cover similar activities for licensees.	

106	United States of America	Article 9	page 113-114	A description of elements required to licensees to comply with their obligations is provided, including compliance with regulations and terms and conditions of the license, personnel training and qualification and openness and transparency. Is there any requirement in the US for the licensee to develop and maintain a management system, including the mentioned elements and others to comply with their obligations for safety?.	Each licensee is required to develop and implement a quality assurance program which comply with the requirements in Appendix B to 10 CFR Part 50 – "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." This program shall be documented by written policies, procedures, or instructions and shall be carried out throughout plant life in accordance with those policies, procedures, or instructions. The applicant shall identify the structures, systems, and components to be covered by the quality assurance program and the major organizations participating in the program, together with the designated functions of these organizations. The quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety. While the requirements of GS-R-3 cover management systems for regulatory bodies, the requirements of Appendix B to 10 CFR Part 50 cover similar activities for licensees.	
101	Russian Federation	Article 10	10.2.	How is safety culture implemented at Rostechnadzor? Which are the main indicators that are controlled by the Regulatory Body in order to check this implementation? Which area is managing this implementation?	There is a system of safety culture indicators in Rostechnadzor based on the documents of OECD NEA "The Safety Culture of an Effective Nuclear Regulatory Body" and the IAEA "The Safety Culture Self-Assessment". The powers, key competences and area of responsibility for each employee are stipulated in the job descriptions and controlled by Rostechnadzor Office for state service and personnel. The Office for state service and personnel performs training, knowledge management and oversight of compliance with the Code of ethics.	

107	Russian Federation	Article 11	11.3.	Could you please explain how is planned and managed retirement of senior experts and how their knowledge is transferred to the next generation of experts within the operating organization?	The forecast of the employees' retirement due to achievement of the retirement age is elaborated annually and used as a basis for recruitment of the graduates. The buddy system is implemented in Rosenergoatom. The main type of the buddy system performance is mentoring, i.e. training of the employees included in the management talent pool. The career and succession management process is implemented in Rosenergoatom, the succession plans are developed. The employees included into the succession plan are trained for th target job positions including in the format of probations. Therefore, the system of critically important knowledge preservation is established to ensure the knowledge succession between the generations. The library of training and methodological materials has been established, the knowledge management system is operable.	

130	France	Article 12	page 98	In the report it is mentioned that: “Following the assessments conducted during the stress tests…ASN has set up a pluralistic working group on these subjects called CoFSOH (Social, organizational and human factors steering committee)…. Since 2012 ….work is done by thematic working groups: …. the interface between “managed safety” and “regulated safety””. Please, could you elaborate on this issue, with some additional information: 1) Rationality behind, and objectives of, the working group on the interface between “managed safety” and “regulated safety”, 2) Links to publicly available documents produced by this thematic working group, and 3) Changes in regulatory practices as a consequence of the work conducted by the CoFSOH steering committee.	ASN considers that there is a need to move forward with regard to the reflections and work being done on the human contribution and organizations to the safety of nuclear facilities and in 2012 it therefore decided to set up the Steering Committee for Social, Organizational and Human Factors (COFSOH), chaired by Pierre-Franck Chevet, ASN’s President. It is a pluralistic working group, which includes ASN members, representatives of institutions and environmental protection associations, personalities chosen for their scientific, technical, economic, social expertise, persons in charge of nuclear activities, representative of nuclear industry professional federations and representative employees’ unions. Since the beginning of 2013 and in parallel with the plenary meetings, the work of the COFSOH has been continuing through four working groups. The forty meetings held to date have addressed the following subjects: (1) subcontracting in normal operating situations, (2) management of emergency situations, (3) interaction between managed safety and regulated safety and (4) legal questions raised in connection with the subjects. The aim of the COFSOH is (i) to allow exchanges between the stakeholders on this difficult topic which are the human and organizational factors and (ii) to write some documents offering common propositions of the different COFSOH members on a given subject. At this time, one document of the GT 1 is public and	

60	Switzerland	Article 12	page 64	In the report it is mentioned that: “The Nuclear Energy Ordinance states that all NPPs must appoint a committee to analyse events and outcomes attributable to human and organizational factors. All NPPs have appointed such committees, who receive adequate education and training on a regular basis”. Please, could you elaborate on this issue, with some additional information: 1) When this part of the Nuclear Energy Ordinance came into force? 2) Are there human and organizational factors specialist on such committees?, 3) Rationality behind the requirement to create such committees focused on events attributable to human and organizational factors, 4) Are there any database at a national level gathering, integrating and assessing such information?	1) This part of the NEO came into force in 2004. Right after the new Swiss Nuclear Energy Act was put into force in 2003). 2) The guideline G07 “Organisation of Nuclear Power Installations” stipulates that a specialist in work and organisational science must be a member of this committee. Therefore one of the member of each of these committees is a person with either a degree in psychology or a degree in engineering in addition with advanced studies in human and/or organisational sciences. 3) Rationality behind this requirement: A nuclear power installation is understood as a socio-technical system consisting of the three components humans, technology and organisation. Therefore, e.g. in the case of an event human, technological and organisational aspect that contributed to the event need to be analysed. The committee’s task is to examine whether the attributable human and organisational factors are adequately analysed. 4) There does not exist any database at a national level. However each nuclear power plant has its own database where the technological as well as human and organisational aspects that contributed to events are gathered.	

61	Switzerland	Article 12	page 64	In the report it is mentioned that, related to Fukushima accident, the Inspectorate has recently published a new report, in 2015, also focused in the field of the human and organizational factors that took place in the accident (in German and to be published in English). Please, could you elaborate on this issue, with some additional information: 1) Are there in that report organizational factors considerations (at the licenses level, at the utilities level, at the regulatory body level, at the government level and at the society level) to many of the Fukushima lessons learned? If yes, please, explain. 2) Link to the English version when publicly available.	The report published in 2015 is the first in a series of reports aimed at deepening the analysis of the human and organisational factors in the Fukushima accident. This first report is descriptive in its nature. It gives an overview of the events and focuses particularly on the description of the main organisations involved in the event response: the Government's and Tepco's Emergency Response Centers based in Tokyo, the organisations located in Fukushima Prefecture, as well as the organisations at the Fukushima Daiichi site. For the latter, staffing and organisation are described. The English translation of the report is under preparation and will be published on ENSI's website. The second part of the report, which is in preparation, will be descriptive as well, with the focus on a rather detailed chronology of the decisions and actions of the staff at the site of Fukushima Daiichi and on the extremely harsh working conditions and countless difficulties they faced while the accident was unfolding during the first days. The last part of the report will be devoted to a reflection on human and organisational factors of the accident in search of possible additional insights for organisations which may be involved in responding to a major event in future.	

113	United Kingdom	Article 12	page 97-98	In the report it is mentioned that: “Another important aspect of ONR’s strategy on leadership and management for safety is the corporate inspection function…… Corporate inspectors are in place for all power reactor licensees”. Please, could you elaborate on this issue, with some additional information: 1) Qualification and training of the ONR’s inspectors acting as corporate inspectors of the licensees, 2) Rulemaking, governing documents and process for conducting such corporate inspections and 3) Recent experience and, when publicly available, links to corporate inspection reports	In the UK there is only one licensee (EdF Energy Nuclear Generation Limited) which operates a fleet of nuclear power reactors across its seven licensed sites. (i) Within ONR the corporate inspection of EdF NGL is undertaken by a lead Corporate Inspector with other specialist inspectors providing additional support. The general qualifications and training requirements of a Corporate Inspector and specialists are the same as those required by all warranted inspectors within ONR. Normally, an ONR Site Inspector with several years’ regulatory experience, with a background in Leadership and Management for Safety (LMfS), is appointed to the corporate inspector role. (ii) The corporate inspector’s interventions are conducted at EdF’s main central office, which is separately located to provide cross-fleet functions to all of its operating reactor sites. The Corporate inspection function involves carrying out fleet-wide inspections of issues that are common across the operating reactor fleet and includes the licensee’s management systems, governance, and cross-fleet learning from experience etc. However, these inspections are carried out in accordance with ONR’s technical Inspection Guides (TIGs) that are published on ONR’s website.	

114	United Kingdom	Article 12	page 98	In the report it is mentioned that: “ONR’s corporate discipline group on leadership and management for safety is well established…. Current areas of focus for the ONR corporate discipline group include: nuclear safety governance (taking into account the lessons from the financial sector on failure of governance processes)…”. Please, could you provide some additional information on: 1) Background on the lessons that could be taken into account from the financial sector on failure of nuclear safety governance processes, and 2) Rulemaking, governing documents and processes considered by ONR to oversee licenses nuclear safety governance	1. ONR takes into account learning and good practice from the financial and other sectors when setting expectations for leadership and management for safety (including governance) in the nuclear sector. Recent examples include: • UK Financial Reporting Council report on corporate culture and the role of boards (highlights good governance as an enabler to a healthy culture) https://www.frc.org.uk/Our-Work/Publications/Corporate-Governance/Corporate-Culture-and-the-Role-of-Boards-Report-o.pdf • UK Chartered Institute for Personnel Development research on ethics in business to inform its ‘profession for the future’ strategy (considers principles-based approach to corporate governance) https://www.cipd.co.uk/Images/best-good-practice-hr-developing-principles-profession_tcm18-8731.pdf • UK Crossrail project learning legacy website (includes lessons learned on project governance) http://learninglegacy.crossrail.co.uk/documents/lessons-learned-from-structuring-and-governance-arrangements-perspectives-at-the-construction-stage-of-crossrail/ Lessons from the above sources will be considered by ONR in the next review of its published guidance on this topic (see below). 2. ONR has set expectations for leadership and	

154	United States of America	Article 12	page 148	In the report it is mentioned that: “The NRC has been processing a few industry requests to transfer operating licenses due to changes of ownership of nuclear power plants”. Please, could you elaborate on this issue, with some additional information: 1) Technical bases for the potential impact of changes of ownership on nuclear power plants safety, 2) Rulemaking, governing documents and process and 3) Recent experience and, when publicly available, links to safety evaluation reports.	The provisions of Section 184 of the Atomic Energy Act of 1954, as amended, and the Nuclear Regulatory Commission’s (NRC’s) regulations at Title 10 of the Code of Federal Regulations (10 CFR) 50.80, “Transfer of licenses,” stipulate that NRC approval is required for transfer of control of the ownership and/or operating authority responsibilities within the facility operating license. Specifically, 10 CFR 50.80(a) states that “no license for a production or utilization facility, or any right thereunder, shall be transferred, assigned, or in any manner disposed of, either voluntarily or involuntarily, directly or indirectly, through transfer of control of the license to any person, unless the Commission shall give its consent in writing.” (https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0080.html) Transfer requests can include either “direct” transfers, which are generally those that involve transfer of ownership or operating authority of the plant itself from one entity to another (e.g., the sale of a plant), or “indirect” transfers, which generally involve transfers of ownership or control of the licensee itself rather than the facility (e.g., the formation of a new parent holding company above a licensee). An application for transfer of a license is required by 10 CFR 50.80(b) to include as much of the technical and	

155	United States of America	Article 12	page 148	In the sub-article 12.4, Fukushima Lessons Learned, it is mentioned that "There are human factors considerations to many of the Fukushima lessons learned". Please, could you provide some additional information on: 1) The role played (and the reasoning supporting that role) by NRC human factors specialists on the Fukushima accident assessment, on the orders issued and on the assessments and inspections of the US nuclear facilities improvement plans, and 2) Are there organizational factors considerations (at the licenses level, at the utilities level, at the regulatory body level, at the government level and at the society level) to many of the Fukushima lessons learned? If yes, please, explain.	1) A senior level task force (referred to as the "Near-Term Task Force," or NTTF) was established at the NRC following the events at Fukushima in 2011. The NTTF developed a set of recommendations, which led to the NRC issuing, among other items, Order EA-12-049, "Order Modifying Licenses with Regard to requirements for Mitigation Strategies for Beyond-Design-Basis External Events." (ADAMS Accession No. ML12054A736) In developing its recommendations, NTTF benefitted from insights from a broad range of NRC experts, including Human Factors Engineering (HFE) and Operator Licensing specialists. Further, HFE specialists also participated in the development of the Mitigation of Beyond-Design-Basis Events (MBDBE) proposed rule, which was published in the Federal Register at 80 FR 70609 on November 13, 2016. In particular, HFE specialists emphasized the importance of including the requirements for an integrated response capability, which would require the integration of beyond-design-basis events response capabilities with the emergency operating procedures, staffing, and supporting organizational structure requirements. HFE specialists further supported the NRC staff during the development of the Japan Lessons-Learned Division Interim Staff Guidance (JLD-ISG)-2012-01, "Compliance with Order EA-12-049, 'Order Modifying Licenses with Regard to Requirements for Mitigating Strategies for Beyond-	

64	Belgium	Article 13	page 68	Quality Assurance Have you regulation for elements important to safety, yet non safety-relate.? If not how do you regulated? Are those elements listed in the Q-List of the NPP´s with any indication o requirement?	In Belgium, we do not use this distinction: only SSCs “importants pour la sûreté” – sometimes translated as “safety related”, sometimes as “important for safety” are defined. The regulation (SRNI-2011)asks for their classification : “All structures, systems and components important to safety, including Instrumentation & Control software, shall be identified and classified according to their importance for safety”. For the new ultimate additional means installed after the Stress Tests, a new specific class has been defined, with specific requirements associated to this new class. These requirements have been discussed with the safety authorities. The Q-list gives an overview of the classification and required qualification level of all safety related SSC's installed on site.	

137	France	Article 13	page 103-104	Could you explain how ASN controls the prior contractor qualification implemented by EDF? There is a standard that specifies for each element or activity its importance to safety and the required quality? How do you verify the effectiveness of the supply chains? Have you implemented tools to address counterfeit and fraudulent items in nuclear facilities? Just in case, please describe them.	The QA/QM system of manufacturers of nuclear pressure equipment of level N1 is assessed under the Module H of the EU Directive 2014/68/EU. This Module enable to evaluate how the manufacturer controls its suplly chain and how efficient is this control. The regulatory framework for subcontracting was strengthened by the decree of 28 June 2016. This decree now limits the number of subcontracting levels to 3, with the objective of guaranteeing the mastery of the activities that have been subtracted by the authorized operator. The French nuclear regulation makes the licensee responsible for controlling their contractors. Therefore, ASN does not inspect directly contractors but regularly inspects the conditions governing the use of subcontracting, both at EDF's suppliers and at nuclear power plants. ASN is currently initiating a reflection to adapt inspection practices by the authority, by the licensee and by the manufacturer in order to adress quality issues and to detect CFSI.	

90	Germany	Article 13	page 97	Posted by Spain It is said that “On the basis of findings obtained the Land authority verifies the effective implementation of the QA systems”: Does this affirmation imply a systematic approach of all kind of Non conformances in each plant? That is: there exists a Corrective Actions Program similar to the ones on USA plants? Have you implemented tools to address counterfeit and fraudulent items in nuclear facilities? Just in case, please describe them.	Within the scope of supervision, there are instruments which are intended to detect accidental faulty actions or unintentional deviations. These instruments include: • 4-eyes principle • Supervision of the work preparation and acceptance process • Access to documents and logs • Check input; Comparison of the ordered with the delivered quality • Independent test procedures operator-expert-authority • Within the scope of random sample supervision, the perception of operator responsibility for safe plant operation is to be strengthened. These instruments are intended to detect deviations irrespective of their condition. The nuclear regulatory framework provides for high demands on production, production monitoring and input testing. All contractors and their subcontractors must be certified according to the German nuclear safety standard KTA 1401. Audits are carried out regularly by the operators (every three years). In the context of goods receipt, the documentation and quality of the delivered goods is also checked as part of a defined QA process. A disqualification of a supplier is possible in case of any abnormalities in the quality	

112	Japan	Article 13	page 88	Could you please enumerate the sections of the Quality Assurance Plan that the licensee submit in the cases of design, manufacturing and services. ¿How do you regulate this plans?	NRA confirms that quality assurance plan and quality management system are appropriately stiputated in Operational Safety Program and licensee's operational safety activity including procurement is appropriately performed through Operational Safety Inspection and Investigation. Regarding Construction Plan or inspections, NRA confirms that licensee's quality assurance plan complies with requirements of NRA Orinance on Quality Management Method, and design related to Construction Plan, plan of construction and inspection are developed based on the quality assurance plan, through the review of Construction Plan.	
113	Japan	Article 13	page 85	How do you define “important to nuclear safety”? And which is the grading approach from the point of view of Quality Assurance for the structures, systems, components and spare parts in function of this definition? Do you have different levels of requirements of QA established in your regulation?	- The NRA Ordinance on Standards for the Location, etc., Article 12 (safety facilities) requires that the safety feature is secured according to the importance of the safety function, and the application of a graded approach is required in the interpretation of the NRA Ordinance. - Regarding important safety facility, SSCs that has functions classified as MS-1 in the safety importance classification indicator such as emergency shutdown of reactor, maintain subcriticality, overpressurisation of pressure boundary, heat removal, core cooling, contain radioactive material.	

71	Sweden	Article 13	page 115, 116, 117	Have Sweden´s NPPs a corrective actions program? Just in case, how is the corrective actions program in Sweeden´s NPPs?	Description of the application of Corrective Action Programmes at the Swedish NPPs is available in chapter 19 of the National Report and in the subchapters as below: 19.2.9 Operating experience feedback function at Ringhals 19.2.10 Operating experience feedback function at Forsmark 19.2.11 Operating experience feedback function at Oskarshamn	
72	Sweden	Article 13	page 115	Which are the nuclear quality standards used to defined the quality requirements?	Quality requirements are governed mainly by Swedish Radiation Safety Authority's (SSM's) regulations, in particular SSMFS 2008:1. Other standards which are used for defining additional safety requirements are for example: • IAEA GS-R-3, GS-G-3.1, • ISO9001 • OHSAS 18001 • US 10CFR50 Appendix B	
66	Switzerland	Article 13	page 65	It is said that as a result of the performance of management system inspections based on the topics of Procurement/Costumer Capability and Competency management has been identified best practices. Could you please send us information about these practices?	The main best practice identified was that every NPP should be aware of its key suppliers with respect to the Business Continuity Management. To guarantee the availability and high quality of products these supplier should be monitored closely. It might be reasonable to tie key supplier in a strategic development partnership. Swiss NNP's exchange about supplier issues in a dedicated working group.	

129	United Kingdom	Article 13	page 102	It is possible to send us, or at least have some details of the reference 33 “Supply chain management arrangements for the procurement of nuclear safety related items or services”? Does this document take into account some methodology to detect Non Conformance, counterfeit, fraudulent and suspect items (NCFSI)?	ONR’s TAG NS-TAST-GD-077 ‘Supply Chain Management arrangements for the procurement of nuclear safety related items or services’ is available on the ONR website. http://www.onr.org.uk/operational/tech_asst_guides/ns-tast-gd-077.pdf This TAG provides a section on Counterfeit Fraudulent and Suspect Items (CFSI) which gives a non-exhaustive list of mitigating measures which could be deployed as part of a purchaser/supplier’s management system as levels of defence against CFSIs for high risk items or services.	
171	United States of America	Article 13	13.4	page 152 Which are the criteria to implement supplemental QA Inspections out of baseline inspection program? How many of this supplemental QA inspections had been performed during the last two years? The pursuit of them are always the same QA criteria or the focus varies?	As described in Inspection Manual Chapter 2515, Appendix B, “Supplemental Inspection Program,” the NRC performs supplemental inspections above the baseline inspections when licensees have one or more inspection findings or performance indicators that exceed the “Green” band (see https://www.nrc.gov/docs/ML1520/ML15204A007.pdf). Quality assurance is not the only aspect covered by supplemental inspections, but a wide range of nuclear safety aspects are also addressed. Supplemental inspections will typically focus on the following quality assurance criteria: organization, design control, procedures, corrective action, and audits. The NRC conducted 39 supplemental inspections in 2015 and 2016. The focus/emphasis of the quality assurance elements may alter depending on the issues observed at the licensee’s facility.	

172	United States of America	Article 13	13.4	page 152 How do you regulate the “augmented quality control” of elements important to safety, yet non safety-relate. Have you regulation for those elements? If not how do you regulated? Are those elements listed in the Q-List of the NPP´s with any indication o requirement. Do you inspect with an specific procedure how has been implemented this “augmented quality control”?	In order to meet some NRC regulations, such as 10 CFR 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants,” licensees may utilize equipment that is non-safety-related to meet those regulations, In such cases, 10 CFR Part 50, Appendix B, would not apply to this equipment since it is non-safety-related, but the associated NRC regulation may address quality aspects. For instance, if a licensee installs an ATWS mitigation system to meet the requirements of 10 CFR 50.62, it is required to “perform its function in a reliable manner.” To address this reliability aspect, and hence quality, the NRC issued Generic Letter 85-06, “Quality Assurance Guidance For ATWS Equipment That Is Not Safety-Related,” to address the “augmented quality” of such equipment (see https://www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/1985/gl85006.pdf). In general, portions of 10 CFR Part 50, Appendix B, are included in that guidance. Since that time, many licensees have incorporated augmented quality assurance criteria into their quality assurance programs similar to the guidance in the generic letter. If the NRC finds an issue with the quality of such non-safety-related equipment, it would need to address the quality issue through the specific regulation associated with that non-safety-related equipment. Licensees are not required to include such	

173	United States of America	Article 13	page 152	Which are the criteria to implement supplemental QA Inspections out of baseline inspection program? How many of this supplemental QA inspections had been performed during the last two years? The pursuit of them are always the same QA criteria or the focus varies?	As described in the Inspection Manual Chapter 2515, Appendix B, “Supplemental Inspection Program,” the NRC performs supplemental inspections above the baseline inspections when licensees have one or more inspection findings or performance indicators that exceed the “Green” band (see https://www.nrc.gov/docs/ML1520/ML15204A007.pdf). Quality assurance is not the only aspect covered by supplemental inspections, but a wide range of nuclear safety aspects are also addressed. Supplemental inspections will typically focus on the following quality assurance criteria: organization, design control, procedures, corrective action, and audits. The NRC conducted 39 supplemental inspections in 2015 and 2016. The focus/emphasis of the quality assurance elements may alter depending on the issues observed at the licensee’s facility.	

174	United States of America	Article 13	page 152	How do you regulate the “augmented quality control” of elements important to safety, yet non safety-relate. Have you regulation for those elements? If not how do you regulated? Are those elements listed in the Q-List of the NPP´s with any indication o requirement. Do you inspect with an specific procedure how has been implemented this “augmented quality control”?	In order to meet some NRC regulations, such as 10 CFR 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants,” licensees may utilize equipment that is non-safety-related to meet those regulations, In such cases, 10 CFR Part 50, Appendix B, would not apply to this equipment since it is non-safety-related, but the associated NRC regulation may address quality aspects. For instance, if a licensee installs an ATWS mitigation system to meet the requirements of 10 CFR 50.62, it is required to “perform its function in a reliable manner.” To address this reliability aspect, and hence quality, the NRC issued Generic Letter 85-06, “Quality Assurance Guidance For ATWS Equipment That Is Not Safety-Related,” to address the “augmented quality” of such equipment (see https://www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/1985/gl85006.pdf). In general, portions of 10 CFR Part 50, Appendix B, are included in that guidance. Since that time, many licensees have incorporated augmented quality assurance criteria into their quality assurance programs similar to the guidance in the generic letter. If the NRC finds an issue with the quality of such non-safety-related equipment, it would need to address the quality issue through the specific regulation associated with that non-safety-related equipment. Licensees are not required to include such	
50	Brazil	Article 14	PAGE 105	This section says: It is noteworthy that the evaluations, studies and implementation made after Fukushima event were widely considered along the holding of the second RPS Angra 1. Related lessons learned from Fukushima events, witch safety improvements for to beyond-design-basis natural hazards has been implemented at Angra 1 NPP?	The safety improvements implemented in Angra 1 resulting from evaluation of BDB natural hazards are discussed in Part D of the Brazilian National report.	

51	Brazil	Article 14	page 110	This section says: The 13 Safety Factors (SF) of the NS-G-2.10 guide have been assessed, as for the Angra 1 PSR, plus an additional one, Severe Accident Management, included as a consequence of the lessons learned from the Fukushima accident. This work resulted in 33 individual assessment reports and one final PSR report containing the summary of the assessments and the Plant global evaluation. Related lessons learned from Fukushima events, witch safety improvements for to beyond-design-basis natural hazards has been implemented at Angra 2 NPP?	The safety improvements implemented in Angra 2 resulting from evaluation of BDB natural hazards are discussed in Part D of the Brazilian National report.	
52	Brazil	Article 14	page 114	This section says: The Regulatory technical activities related to nuclear power plants and research reactors licensing are carried out by the CGRC,…. ………… Supervises the operation of nuclear installations, analyzing eventual technical modifications; ………… How are the conditions under which licensees may make changes to the facility or procedures and conduct tests or experiments without prior Regulatory Body approval?	All the modifications that don’t impact the Safety Analysis can be done by the operator without previous approval from CGRC or CNEN. In others words, a modification has to be approved by the regulatory body if: 1 – increase the probability of an accident or upset operation or its consequences ; 2 – create a new accident or upset conditions; 3 - reduce the safety margins stablished in the safety analysis. In case of Research Reactors a similar process is applied, if the tests or experiments that will be performed don’t impact the Safety Analysis, it can be done by the operator without previous approval from CGRC or CNEN.	
88	Finland	Article 14	page 59	Assessment and verification of safety Knowledge Management is identified as a challenge for licensees. • Is there in Finland any regulatory guidance on this issue?	There is no specific guidance in how to implement the Knowledge Management but there are YVL requirements concerning Knowledge Management. E.g. YVL A.4 requirement 319. The licensee shall ensure that knowledge and competence are duly shared; the atmosphere prevailing in the organisation shall promote such sharing and effective procedures are in place to support sharing.	

89	Finland	Article 14	page 6 and 50	• How has PSA been used during PSR to decide on the modernization projects to be undertaken? • Do STUK Guides provide criteria to decide on this regard? • Is there any definition by the regulator of PSR evaluation criteria in STUK Guides or elsewhere?	•PSA has been used to identify needs for plant modification and in the comparison of possible alternative modifications and their effectiveness. In general,decisions on modifications are not associated only with the PSRs but they are rather implemented when needs are identified. In modernization projects not related to safety improvements, eg. power uprates, PSA is used to ensure that there is no significant increase of risk. • STUK's YVL Guides include the general requirement tht PSA shall be used in the identification of needs for safety improvemnnets and evaluation of plant modification but do not provide detailed criteria on this issue. • The evaluation criteria in PSR are the same as for the renewal of the operating licence application. Guidance on the operating licence application and PSR is given in the Guide YVL A.1 issued by STUK.	
90	Finland	Article 14	page 54	Verification of safety STUK Regulation (STUK Y/1/2016) includes several requirements which concern the verification of the physical state of a nuclear power plant. ……. Main programmes used for verification of the state of a nuclear power plant are • periodic testing according to the Operational Limits and Conditions • maintenance programme • in-service inspection programmes for pressure retaining components • surveillance programme of reactor pressure vessel material • research programmes for evaluating the ageing of components and materials. Which are the conditions under which licensees may make changes to the facility or procedures and conduct tests or experiments without prior Regulatory Body approval for have reasonable assurance that plants continue to conform to the licensing basis?.	Licensees may make changes without prior approval to SSCs' routine maintenance programmes based on their gathered field experience. However, even such programmes are to be provided at STUK's disposal and reviewed by STUK when necessary. Changes of inspections and tests within Operational Limits and Conditions or in-service inspections of pressure retaining components may be proposed but changes are subject to STUK's approval before they can be implemented.	

147	France	Article 14	page 109	This section says: In accordance with the principle of continuous improvement of reactor safety levels, but also to improve the industrial performance of its production tool, the licensees periodically made modifications to the equipment and the operating rules. These modifications are for instance the result of processing of deviations, periodic safety reviews or the integration of operating experience feedback. The BNI procedures decree defines the requirements concerning the implementation of changes by the licensees and their review by ASN. The procedures for managing and notifying hardware modifications were specified in ASN resolution 2014-DC-420 of 13/02/2014. Explain assessment process that holders have to carry out to determine if a change in design or in operating rule modify the criteria, standards and conditions in which the authorization is based (may affect safety) and in which cases these changes require approval of the ASN	The process implemented depends on the impact’s significance of the change on the protected interests, including safety, defined by the BNI decree. The first type of process is related to "substantial" modifications and is already describe in Section 7.2.9 of the ASN report. The second type of process is related to "significant" modifications when they affect the facility's safety report or impact assessment content. Depending on their relevance, the significant modifications are submitted either to notification to ASN or to authorization by this authority. The criteria for selecting between notification and authorization procedures are due to be defined by an ASN decision by the end of 2017. In the meantime, all significant modification are submitted to authorization. The third type of process includes the other modifications than those aforementioned. Their management is defined in the licensees' internal process, and are not subject to administrative procedure.	

148	France	Article 14	page 107	This section says: In addition to the procedures applicable to changes to the installations or their operating mode, the Environment Code requires that the licensee carry out a periodic safety review of its installation every 10 Does the periodic safety review of its installation every 10 years, follow the recommendations (scope and criteria) of IAEA Safety Guide SSG-25 (2013)? If the scope or criteria of the RPS are different to SSG-25, explain the differences	In France, the scope of the PSR is similar to the scope described in section 2.9 of the guide SSG-25 (required by the environment code - article L. 593-18 an L.593-19). Moreover, the periodic safety review in France takes into account the recommendations of WENRA (for example, the recommendations of WENRA 2014 will take into account for VD4-900 PSR) and includes the assessment of environmental consequences due to non radiological risks and the drawbacks resulting from normal operation of the facility.	
149	France	Article 14	page 111	This section says: The safety review of the reactors, carried out by means of periodic safety reviews or reviews of particular thematics, leads in a certain number of cases to nuclear reactor modifications. In most cases, these modifications are made in batches, each batch being implemented on all the reactors of the plant series concerned, with an initial reactor, referred to as the "first off", playing the role of prototype. This grouping of modifications allows greater consistency and industrialisation by facilitating scheduling, documentation updates and operator training. These batches are generally implemented during the ten-yearly outages in order to minimise the impact of the work on reactor availability. Explain briefly some examples of improvements most important implemented in nuclear power plants derivatives from Periodic safety review	Please refer to section 6.3.1.1 and its subsections of the Report (p. 37-43).	

96	Germany	Article 14	page 101	This section says: Safety assessments are also submitted to the supervisory authority in the course of licence applications for modifications of the plant or its operation pursuant to § 7 of the AtG or modifications subject to approval within the framework of supervision according to § 19 of the AtG. The licensing procedure for modifications pursuant to § 7 of the AtG is basically performed according to the same regulations described above for the granting of a construction licence. This also applies to the documents to be submitted and the safety assessment based on them (? Article 7 (2ii)). As regards modifications of the nuclear installation or its operation that are not subject to licensing The modifications of the plant could have different causes and objectives (for example: fixing a problem, to improve the operation of a safety system, update or renew the technology, etc). Which criteria are used to decide whether a modification of the plant is implemented or not in a NPP whose closure is expected in few years?	Generally, the criteria used to decide whether a given modification of a NPP which is scheduled to be closed in a few years has to be implemented or not are independent of the residual lifetime. Until the final day of operation, the necessary precautions against damages – in the light of the state of the art in science and technology – have to be taken. Further, § 7d of the Atomic Energy Act requires the operator to implement those measures that will improve nuclear safety unless they would only contribute to a minor risk reduction. The regulator will check whether such measures are proportionate concerning the required time of technical implementation and the prospected time of its effectiveness.	

97	Germany	Article 14	table 14-1 page 103	Notes under the tableTable 14-1 says: Shaded fields denote the nuclear installations that have been shut down. * Safety review performed, no evaluation ** No future safety review required according to § 19a para. 2 AtG (Power operation will cease no later than three years after the ten-year review interval). Apparently, Grafenrheinfeld (KKG) and Gundremmingen B (KRB B) do not correspond with any of the notes in the table 14-1. Have been these nuclear installations shut down or will cease no later than three years after the ten-year review interval)?	The Grafenrheinfeld NPP (KKG) was shut down in June 2015 (see page 35 of the National Report, the line for KKG in table 14-1 has to be shaded, thank you for remarking the error) and as such does not require a safety review. The Gundremmingen B NPP (KRB B) will be shut down by the end of 2017 (see page 44 of the German report). According § 19a (2) AtG : “1The obligation to submit the results of a safety review and evaluation shall not apply if the licensee gives a binding declaration to the supervisory authority and the licensing authority stating that operation of the installation will be permanently discontinued no later than three years after the dates specified in Appendix 4. … 3The authorisation to operate the installation shall expire as per the date cited in the owner’s statement pursuant to sentence 1. Sentences 1 and 2 shall apply accordingly in the event of para. (1), sentence 3.” no more periodic safety review will be required. Thus, both plants have a shutdown date which is before the next scheduled 10-year safety review).	

98	Germany	Article 14	page 101	This section says: Safety assessments are also submitted to the supervisory authority in the course of licence applications for modifications of the plant or its operation pursuant to § 7 of the AtG or modifications subject to approval within the framework of supervision according to § 19 of the AtG. The licensing procedure for modifications pursuant to § 7 of the AtG is basically performed according to the same regulations described above for the granting of a construction licence. This also applies to the documents to be submitted and the safety assessment based on them (? Article 7 (2ii)). As regards modifications of the nuclear installation or its operation that are not subject to licensing What criteria are applied to determine if a change in design or document of the plant are subject to licensing?	Regarding changes and/or modifications (technical, structural or administrative), a distinction is made between "major modifications" (approval, licence) and "minor modifications" (supervision). The criteria for distinguishing a “major” from a “minor” modification are specified in the valid operating regulations regarding the procedure for maintenance / modification measures. Major modifications are e.g. those with whom • the technical protection objectives of the “Safety Requirements for NPPs” of the BMUB are affected, • the underlying accident spectrum is changed, • the basic technical solutions to with which the protection objectives are adhered to in the case of the accident spectrum. The requirement for a modification may arise, among other things, from the need to adapt the plant according to the requirement to take the necessary precautions against damages in the light of the state of the art in science and technology.	
99	Germany	Article 14	page 102	This section says: **No future safety review is required to Grohnde (KWG) (PWR), Philippsburg 2 (KKP 2) (PWR) and Isar 2 (KKI 2) (PWR) because power operation will cease no later than three years after the ten-year review interval). Will be applied some kind of security review to these three stations (partial RPS) for the additional years after the required ten year review interval	Periodic security reviews and periodic safety reviews have to be conducted simultaneously under the same rules. The licence for decommissioning includes a complete security concept. Every modification of the security concept has to be reviewed and accepted by the supervisory authority.	

131	Russian Federation	Article 14	14.5	Concerning periodic safety reviews, is the Russian regulation Implementing the IAEA SSG-25 Periodic Safety Review for Nuclear Power Plants (published in 2013)?	Russia has a generally similar document in effect at the safety guide level: "Guide for the Periodic Nuclear Unit Safety Assessment" (RB-041-07). It was put into effect on 1 January 2008 and takes into account the experience accumulated in Russia by the time and the IAEA standards developed by then.	
75	Sweden	Article 14	page 119	It is indicated that: "All safety systems as well as other plant structures systems and components of importance for the defence-in-depth shall be described in the SAR: • Is there a common definition of the concept "important for safety" or importance for "defense-in-depth" • Is there a rule, method or guide to set the scope of those type of components in a standardized way	The systems and equipment, additional to safety systems that have an essential importance to the plants defense in depth, such as those with potential impact on fulfillment of safety functions and protection around the plant are included in SAR, based on operating experience and probabilistic safety analyses. In the development of new regulations, SSM's intention is to more closely follow the IAEA recommendations.	

76	Sweden	Article 14	page 127/page 120	This section says: Section 14.1.3. This section that shows two types of review are contemplated: the primary review, shall be carried out within those parts of the licensee's organisation which are responsible for the specific issues. The second step, the independent review, shall be carried out by a safety review function (a safety committee), established for this purpose and with an independent position in relation to the organisation responsible for the specific issues. This section says: Section 14.2.7 Safety reviews This section describes three types of reviews: First, a primary review is carried out by the operations department, that is primarily responsible for reactor safety. If needed, resources from other departments are utilized. A second, independent, review is then performed by an independent department or function within the licensee organisation. This independent department (10–15 experienced engineers) or function shall not been involved in the preparation or execution of the issues under review. A third type of review is performed by the safety review committees and councils at different levels of the power plant organization Explain the relationship between revisions described in section 14.1.3 Verification of safety decisions and Safety	Section 14.1.3 describes the requirements by SSM, which are a primary review and a second independent review by a safety committee. Section 14.2.7 describes implementation of the requirements by a licensee. The procedure of the licensee sets up a process with an additional review to the ones described in 14.1.3. The phrase "second independent review" is here used in a different sense than in 14.1.3. In this licensee procedure, the third review step is presenting the second independent review required by SSM.	

77	Sweden	Article 14	page 127-128	This section 14.2.7 Safety reviews describe three types of reviews: First, a primary review is carried out by the operations department,that is primarily responsible for reactor safety. If needed, resources from other departments are utilized. A second, independent, review is then performed by an independent department or function within the licensee organization. This independent department (10–15 experienced engineers) or function shall not been involved in the preparation or execution of the issues under review. A third type of review is performed by the safety review committees and councils at different levels of the power plant organization How is assured that the results of the review of second independent review y the third review are implemented?	Section 14.1.3 describes the requirements by SSM, which are a primary review and a second independent review by a safety committee. Section 14.2.7 describes implementation of the requirements by a licensee. The procedure of the licensee sets up a process with an additional review to the ones described in 14.1.3. The phrase “second independent review” is here used in a different sense than in 14.1.3. In this licensee procedure, the third review step is presenting the second independent review required by SSM. Regarding SSM’s control of requirements on safety review the following is the case. When the application is submitted to SSM, there is a requirement that the notes from the independent review (safety committee) shall be attached. SSM reviews the application, including these notes. If the SSM reviewers need additional material or information, it will be requested.	

78	Sweden	Article 14	page 120/127	Section sección 14.1.3 says the following: The primary review, shall be carried out within those parts of the licensee's organisation which are responsible for the specific issues. The second step, the independent review, shall be carried out by a safety review function (a safety committee), established for this purpose and with an independent position in relation to the organisation responsible for the specific issues. Section 14.2.7 Safety reviews says the following: First, a primary review is carried out by the operations department, that is primarily responsible for reactor safety. If needed, resources from other departments are utilized. A second, independent, review is then performed by an independent department or function within the licensee organisation. This independent department (10–15 experienced engineers) or function shall not been involved in the preparation or execution of the issues under review. A third type of review is performed by the safety review committees and councils at different levels of the power plant organization What type of monitoring or review makes the Regulatory Body over those three different types of safety reviews performed by the licensee holders?	Section 14.1.3 describes the requirements by SSM, which are a primary review and a second independent review by a safety committee. Section 14.2.7 describes implementation of the requirements by a licensee. The procedure of the licensee sets up a process with an additional review to the ones described in 14.1.3. The phrase "second independent review" is here used in a different sense than in 14.1.3. In this licensee procedure, the third review step is presenting the second independent review required by SSM. Regarding SSM's control of requirements on safety review the following is the case. When the application is submitted to SSM, there is a requirement that the notes from the independent review (safety committee) shall be attached. SSM reviews the application, including these notes. If the SSM reviewers need additional material or information, it will be requested. In addition, SSM controls that required functions for safety reviews are implemented in the licensees' management systems (processes and procedures).	
79	Sweden	Article 14	page 124/125	This section say: The licensees are required to submit a PSR of each reactor unit at least every 10 years. The analyses, assessments and proposed measures as a result of the review shall be submitted to SSM. Typically a project is formed to conduct the review, involving 15-20 staff of the licensee Typically, how many resources from Regulatory Body involve the evaluation of each PSR and how many time spend?	A typical PSR involves about 45 experts. The number of man-days in total for the review varies from 400 to 600. The latest PSR used 476 man-days and about 25 were used for project management.	

80	Sweden	Article 14	page 124/125	This section says: The licensees are required to submit a PSR of each reactor unit at least every 10 years. The analyses, assessments and proposed measures as a result of the review shall be submitted to SSM Could give examples of type of measures has been proposed by licensees, as result of PSR?	Some recent examples are: • Updating of maintenance programme • Time limiting safety analyses of primary systems components • Some improvements coming from stress test results Many other identified measures are related to LTO and action plans are developed.	
69	Switzerland	Article 14	page 24/25	This section say: The following additional points help to ensure that the physical state of an NPP complies with its licence: • Modifications important for safety require a permit granted by the Inspectorate. • A plant review must be carried out after each refuelling outage. • The Inspectorate has an efficient inspection programme in place in order to verify compliance with licensing requirements. Which are the main item and characteristics of the plant review carried out after each refueling outage?	The main items of the plant review while and after each refuelling are • fuel inspection results and fuel physics report, • preliminary technical report of the outage, • component and material tests, • system functioning tests • the startup tests • documentation and • outage final inspections. This review is the basis of the inspectorate decision for the permit of the next cycle.	

70	Switzerland	Article 14	page 64	This section says: For existing plants, a Periodic Safety Review (PSR) is required at least every ten years. Important elements of a PSR are an update of the Safety Analysis Report (SAR), an assessment of design basis accidents, an assessment of the ageing surveillance programme, an update of the Probabilistic Safety Analysis (PSA) and an evaluation of operating experience over the last 10 years. The details (scope and process) of a PSR are defined in the Inspectorate's Guideline ENSI-A03. Are the requirements (scope and criteria) of PSR comparable to those recommended in the IAEA Safety Guide SSG-25 - Periodic Safety Review for Nuclear Power Plants, issued in March 2013? If the scope or criteria of the RPS are different to SSG-25, explain the differences	The Regulatory Guide ENSI-A03 covers the requirements of IAEA Safety Standard SSG-25 „Periodic Safety Review for Nuclear Power Plants“. All 14 safety factors of SSG-25 are covered by ENSI-A03. The main difference is an additional extension of ENSI-A03 in terms of requirements for the review of long term operation.	

132	United Kingdom	Article 14	page 106	Which is the scope of the assessment and verification of safety (Article 14) in terms of SSC (Structures, Systems and Components)? Are also included SSC that, not being "safety-related" could be "important to safety"?	In the UK, the scope of the assessment and verification of structures, systems and components (SSCs) important to safety is subject to the categorisation of safety function(s) that these are intended to perform and the classification assigned to each SSC by the NPP operator. The categorisation and classification is assessed by ONR in accordance with its Safety Assessment Principles (SAPs) ECS1 (Safety categorisation) and ECS2 (Safety classification of SSCs), respectively (refer to Article 18, para 18.67 of the UK report). This aligns with recognised international practice, such as that provided in IAEA TecDoc 1787 and IEC 61226, which is reflected in ONR's Technical Assessment Guide NS-TAST- GD-094 (http://www.onr.org.uk/operational/tech_asst_guides/index.htm). Licence conditions (LCs) issued by ONR to UK NPP operators requires adequate arrangements for the production and assessment of safety cases to be developed and implemented to justify safety throughout the lifecycle of the plant. These arrangements should set-out the methodology for the identification and categorisation of safety functions, the classification of SSCs and how this information should be generated, underpinned and used in the production and assessment of the safety case. The safety case should therefore identify and categorise the necessary safety functions,	

184	United States of America	Article 14	page 173, 174	Immediately after the event, using the existing Reactor Oversight Process, the NRC conducted inspections and issued orders, INs, and bulletins to aid in determining the preparedness of U.S. nuclear power plants to withstand a similar event. Furthermore, the Reactor Oversight Process will be used to assess and verify that changes currently being implemented in response to lessons learned from the accident were completed properly Has the NRC made any estimate of the resources that has devoted to Lessons Learned at Fukushima Dai-ichi events (inspections and issued orders, INs, and bulletins)?	From fiscal year 2012 through fiscal year 2016, the NRC has budgeted approximately $120 million on post-accident inspections, issuing and implementing the orders, issuing the request for information and reviewing the responses, and other related support work. This does not include the billions of dollars spent by the industry enhancing safety in response to the new NRC requirements.	

185	United States of America	Article 14	page 167/168	The controls on generic backfitting include a Committee to Review Generic Requirements review, which is a committee of senior managers from different NRC offices. Established in 1981, this committee operates under a charter that specifically identifies the documents to be reviewed and the analyses, justifications, and findings to be supplied to this committee by the NRC staff. Its objectives include eliminating unnecessary burdens on licensees, reducing radiation exposure to workers while implementing requirements, and optimizing use of NRC and licensee resources to ensure safe operation. Therefore, the Committee to Review Generic Requirements' charter is a key implementing procedure for generic backfitting, although the primary responsibility for proper backfit considerations belongs to the initiating organization. Indicate some specific recent examples of application on optimizing NRC resources to ensure safe operation	The Committee to Review Generic Requirements (CRGR) ensures that proposed generic backfits to be imposed on the U.S. Nuclear Regulatory Commission (NRC)-licensed power reactor, new reactors, or nuclear materials facilities are appropriately justified based on backfit provisions of applicable NRC regulations (i.e., 10 CFR 50.109, 10 CFR 52.39, 10 CFR 52.63, 10 CFR 52.98, 10 CFR 70.76, 10 CFR 72.62, or 10 CFR 76.76) and the guidance contained in the Regulatory Analysis Guidelines (NUREG/BR-0058) (https://www.nrc.gov/reading-rm/doc-collections/nuregs/brochures/br0058/br0058r4.pdf) or the Commission's backfit policy. The CRGR's primary responsibilities are to recommend to NRC's Executive Director for Operations (EDO) either approval or disapproval of the staff proposals and to provide guidance and assistance to the NRC program offices to help them implement the Commission's backfit policy. The backfit regulations contain requirements that the NRC must satisfy to impose backfits on licensee facilities. In general, the NRC has two standards to evaluate when considering modifications. The modification is either required to ensure adequate protection or is cost beneficial. The adequate protection standard establishes the minimum level of public safety that the NRC must maintain. Cost beneficial modifications consider both	

40	Viet Nam	Article 14	page 25/26	This section says: In the Master Plan for the National Energy Development during the period from 2011-2020 with the vision to 2030 (MP No. VII), the Government of Vietnam planned to put the first 2 units (1,000MW each) into operation in 2020 and by 2030, nuclear power is projected to produce 10,700 MW, accounting for 10.1% of the total national capacity. Investigation of 2 sites for the first 2 NPPs was completed. 5 sites for the third NPP were planned for investigation. The first 2 NPPs (Ninh Thuan 1 and 2) with one unit at each site were scheduled to be in operation by 2020-2021. Has the Regulatory Body developed a Management System, including the necessary processes and the corresponding procedures, for the different stages of the licensing of nuclear power plan projected?	The integrated quality management system for the Regulatory Body including the necessary processes and the corresponding procedures, for the different stages of the licensing of nuclear power plan is now under developing. VARANS is lack of experience in developing this management system. To dealt with this difficulty, VARANS get support from EC' experts under the Task 2 "Further development of a quality management system for use by VARANS in the regulation of nuclear installations" of bilateral project INSC VN3.01/13 "Enhancing the capacity and effectiveness of the Vietnam Agency for Radiation and Nuclear Safety and its Technical Support Organizations".	
172	France	Article 14.2	page 108	• To what extent is being used de OIEA SSG-25 guide for the periodic safety reviews in France?	In France, the scope of the PSR is similar to the scope described in section 2.9 of the guide SSG-25 (required by the environment code - article L. 593-18 an L.593-19). Moreover, the periodic safety review in France takes into account the recommendations of WENRA (for example, the recommendations of WENRA 2014 will take into account for VD4-900 PSR) and includes the assessment of environmental consequences due to non radiological risks and the drawbacks resulting from normal operation of the facility.	

149	United Kingdom	Article 14.2	page 110	Which is the scope of the update fof the PSA´s in UK? Level 1 PSA? Level 1 and Level 2 PSA? Others?	The PSAs for all operating reactors within the UK are "living PSAs" and updated approximately every three years, or sooner if there are significant changes to plant or operations that require a more frequent update. The updates include revisions to Initiating Event Frequencies (IEFs), plant reliability data, hazards analysis and other modelling aspects. The pressurised water reactor (PWR) at Sizewell B has a full scope Level 1, 2 and 3 PSA. The Level 1 PSA is updated to provide an estimate of the core damage frequency (CDF) as part of the living PSA programme and this used to provide revised Level 2 and 3 dose / risk information. The PSAs for the Advanced Gas Cooled Reactors (AGRs) are hybrid PSAs and include a Level 1 PSA and elements of a Level 3 PSA in the form of off-site dose estimates to a person in five dose bands (Target 8 of ONR's Safety Assessment Principle (SAPs) (Ref. 1). A Level 2 PSA has been carried out for one AGR that is representative of the fleet. As part of the living PSA programme, the AGR Level 1 PSA is updated in addition to the off-site dose estimates. For new build reactors (for example Hinkley Point C), Level 1, 2 and 3 PSA are / will be carried out consistent	
117	Finland	Article 15	page 63, table 4	Information on the activity of the radioactive effluent is provided in the report: noble gases, iodines and aerosols (airborne effluents) and liquid effluents excluding tritium Please, could you inform if the activity of tritium and C-14 is also measured in the liquid and gaseous effluents? If yes, could you provide information on the activity values?	The nuclear power plants in Finland have a regulatory requirement to measure tritium from liquid and gaseous effluents and C-14 from gaseous effluents. In 2015 the total amount of tritium released to the air was 1,47E11 Bq from Loviisa NPP and 1,04E12 Bq from Olkiluoto NPP. The amount of tritium released to the sea was 1,64E13 Bq from Loviisa NPP and 2,05E12 Bq from Olkiluoto NPP. The total amount of C-14 released to the air was 4,15E11 Bq from Loviisa NPP and 1,07E12 Bq from Olkiluoto NPP.	

94	Sweden	Article 15	page 145	According to the report, the concepts of reference values and target values are used for nuclear power reactors as a measure of the application of BAT for reducing releases of radionuclides, values that are defined by the licenses Please, could you provided additional information on those reference and target values	According to the Swedish Radiation Safety Authority's Regulations on Protection of Human Health and the Environment in connection with Discharges of Radioactive Substances from certain Nuclear Facilities, SSMFS 2008:23, each nuclear power reactor are required to determine the so-called reference values and target values. The reference values should represent a typical value for discharges from a specific reactor during normal operation, and are normally represented by a selection of a few easy-to-measure nuclides as representatives of each category, noble gases, particulates etc. Target values should represent the discharge of separate radioactive substances or groups of radioactive substances and to which levels the discharges could be reduced to in a specified period of time. The intention with target values is that it should be set low enough to be challenging to current performance.	

154	China	Article 16	page 127	Regarding the upgrading renovation and consolidation of nuclear accident emergency commanding center, as one of the improvement actions implemented by NPP after Fukushima accident, to what extent are these emergency commanding centers improved?: Are they newly built? Are they seismic resistant? Can they cope with surrounding air contamination due to radioactivity release? How far from the nuclear reactors are they located?	1) After Fukushima nuclear accident, the anti-seismic requirements on emergency center are as follows: Under the civilian specification system, the anti-seismic design is based on the basic intensity of the code for seismic design of buildings and structures plus I degree. To meet the requirements on habitability of the emergency center under SL-2 condition, elastic design shall be made according to the civilian response spectrum of ground acceleration (not lower than Class II site) equivalent to SL-2. When the emergency center is located in places lower than Class II site, the site soil-layer analysis shall be performed to determine the input acceleration value again. New NPPs in China will be built according to the above anti-seismic requirements. For operating NPPs and NPPs under construction that do not meet the requirements, a standby emergency center will be established. (As new units are built in some nuclear power plants under construction, new emergency centers are built to meet the above requirements and anti-seismic modification is made for structures during the transition period.) 2) The design of emergency center ensures its habitability during radioactive release under severe accident condition, including such design measures as shielding and ventilation filtration. 3) The distance to reactor is generally no more than 2km.	
120	Finland	Article 16	page 67	Regarding the use of the Nordic Flag Book and Nordic Manual that have a broad consensus among Nordic countries, how would they be used in case of an emergency within the Russian territory that would be able to affect Finnish territory, given that Russia has not taken part in developing the above mentioned documents?	The documents would be used to decide and implement protective actions in Finnish (and other Nordic countries') territory based on the expected impact on the those areas, similarly to accident within Nordic Countries. The documents apply whether the accident happens in a Nordic country or outside it. In this kind of case, the Russian authorities would of course follow their protection strategy and communication between the countries would rely on bilateral agreements, but else the documents would be just a usable.	

8	Iceland	Article 16	page 12	Very little information has been provided regarding communication to the public. Could you please elaborate about sharing of responsibilities, coordination among authorities, and coordination with foreign countries in the field of communication the public and media?	Iceland thanks Spain for this question, which is marked to refer to Article 16 (p. 12 in the NR of Iceland) and would like to point out that the topics of the question are addressed in other parts of the report. Openness and transparency are core concepts of the Information Act No. 140/2012, which applies to all operations of IRSA. The objective of this Act is to guarantee transparency in government administration and the handling of public interests, as described on p. 11 of the NR of Iceland. It is the Authority‘s policy to increase the release of information to the public as applicable. The Icelandic population is relatively homogeneous. >96% of homes have Internet connections (2014, highest in Europe with NL and LU), virtually all have telephone, TV and radio and speak the native language and/or English. The civil protection system has become very well established due to the imminent threat of various natural hazards. Ways to communicate urgent information to the public are well established and are tested on a regular basis, in real situations if not in exercises. IRSA works in close cooperation with the Department of	

10	Oman	Article 16	page 27	How large is the scope of the Gulf Cooperation Council (GCC) Regional Radiological and Nuclear Emergency Preparedness and Response Plan? Does it encompass harmonization of protective measurements, harmonization of information to the public, sharing of information prior and during emergency? Has the above mentioned Plan statements to cope with situation when neighboring countries do not consider appropriate the respond of the accident country?	The GCC Regional Radiological and Nuclear Emergency preparedness and Response (RRNEPR) Plan contains all the elements of an emergency plan, as recommended in the IAEA safety standards and guides. The plan addresses: - the planning basis; - the emergency response process harmonized for all GCC Member States, including (i) coordinating information exchange and communication between Member states and taking protective measures, (ii) the required regional response for all the identified threats, (iii) operational intervention levels, etc… ; - emergency preparedness process, including, (i) coordination by the regional emergency response center (the GCC Emergency Management Center in Kuwait), (ii) its required logistical support and facilities, etc… With respect to the question if the Plan "contains any statement coping with situation when neighboring countries do not consider appropriate the response of the accident country", the RRNEPR Plan does contain any such explicit statement. It defines however the overall responsibilities of the regional emergency response center which are, inter alia, to ensure sharing and coordination of resources to prepare and respond to a radiological or a nuclear event and to ensure consistency in the response of the various Member States following a radiological or a nuclear event.	
11	Senegal	Article 16	page 13	Will the national plan of radiological emergency that must be developed by ARSN in collaboration with all relevant national structures be in line with IAEA GSR part 7?		

95	Sweden	Article 16	page 158	It is indicated that a number of exercises are conducted annually related with accident management, communications, environmental monitoring, etc.: • Do the Swedish plants also conduct firefighting drills using the “FLEX” equipment? • Is there any requirement associated to the time needed to deploy the (FLEX) equipment in those cases (big fires)?	No, the Swedish plants are not conducting firefighting drills using the FLEX equipment. However, this does not rule out the possibility for the FLEX equipment to be used for firefighting in case of failure to extinguish fire with other equipment dedicated for the purpose. It should be noted that the FLEX equipment mainly consists of floodlights, portable power units, bilge pumps and mobile diesel generators to secure the power for reactor safety systems. The FLEX equipment is used several times a year at all three power plants during training and drills of various types. The number of occasions and type of training differs somewhat for the different power plants. However, emphasis lies foremost on training to prepare and testing of the equipment for core cooling functions. There are no regulatory requirements, but there are recommended time limits for the equipment to be operational, set by the licensees.	
41	United Arab Emirates	Article 16	page 77	Regarding the on-site emergency planning, do the actions undertook by ENEC to enhance emergency preparedness after Fukushima-Daiichi accident include provisions to store and maintain portable equipment for electrical and water supply?	ENEC’s post Fukushima plans include provisions to store and maintain portable equipment for electrical and water supply, including portable pumps, hoses, and auxiliary equipment, as well as mobile diesel generators. The Barakah Accident Management Programme (AMP) include provisions for mitigation strategies to restore reactor core cooling, containment integrity control, spent fuel pool cooling capabilities using such equipment. As required by FANR Regulation 16 Article (19), such equipment will be located in a safe area, protected against hostile actions and credible external hazards so as to ensure its availability.	

65	Brazil	Article 16.3	page 136	Within the scope of the lessons learned after Fukushima, has the license holder made an assessment of the personnel resources, and other kind of resources, available at the site in order to respond to an emergency when a very severe natural hazard could have affected off-site infrastructures?	Yes, this evaluation has been made, considering temporary impossibility to access the side by road, total loss external power and loss of fresh water supply (disruption of the fresh water supply system): " Access is possible by sea for personnel and supplies; the Brazilian Navy, one of the organizations that take part of the External Emergency Plan, can provide large barges that can dock onto the site, for people and supply transportation; " There is sufficient fuel on site for about one week of operation of the plants emergency DGs; " The mobile equipment is located onsite, at about 1 Km from the plants on a plateau, 40 m above site level, not subjected to the external events that can affect the Plants. Transportation to the Plants can be done through alternate routes. " As an alternative for fresh water supply an additional seismic reservoir (4.000 m3) is to be built in the same plateau where the mobile equipment is located. The design of the reservoir is ready.	
52	Viet Nam	Article 16.3	page 38	Does the national radiological and nuclear emergency response plan in Vietnam, and with regard to nuclear accidents in neighboring countries that can affect Vietnamese territory, consider applying in a very early phase of the accident and within the Vietnamese territory the same protective actions taken by the country where the accident has occurred?	Yes. The criteria for applying urgent protective actions for the early phase of nuclear accident in neighboring countries had been already developed (Circular 25/TT-BKHCN). In NRERP, requirements on urgent protective actions, for instance relocation, sheltering, shall be followed these above criteria. In the near future, these above criteria shall be modified to be comply with updated IAEA guidance.	

102	Switzerland	Article 18	page 105-106	Pages 105-106: ENSI has required an inspection of reactor vessel base material after WENRA recommendation derived from Döel 3 and Tihange 2 findings. Which was the regulation tool (instruction, mandatory letter…) to ask for such inspection? Were specific schedules required or the plants could accommodate the inspection in their normal ISI intervals?	Inspection was required with a mandatory letter based on para. 2 and 3 article 4 of the ordinance on vessels and piping VBRK (SR 732.13) for special testing. ENSI requested the special testing during the next ISI for RPV welds.	
151	Finland	Article 19	page 82	Regarding the Loviisa monitoring programs for the carbon steel piping , which are the main results of these programs in relation to the piping lifetime?	The Loviisa monitoring program is established to control the operability of the secondary pipe lines. Thickness measurements are conducted to find erosion corrosion in the piping and surface inspections are used to detect fatigue cracks. In addition, digital radiography is used to detect corrosion in small pipes (D < 200 mm). Thickness measurements and surface measurements are conducted during annual outage and digital radiography in normal operation phase. The main target of the monitoring program is to prevent adverse effects of ageing mechanisms (erosion corrosion, fatigue and corrosion) on the operability. In addition, these results determine the interval for the repair, modifications and replacement of the secondary pipe components.	
111	Switzerland	Article 19	page 30	Page 30: The safety evaluation report from ENSI on the PSR of each Swiss NPP have been made accessible to public (“publicly available”). Which is the used tool to do this? Internet (which web-site)? Announce for public demand?	Safety evaluation reports from ENSI on the PSR of Swiss NPPs have been published on the internet (www.ensi.ch) .	

250	United States of America	Article 19.4	page 227 paragraph 6	Regarding the proposed rule to develop mitigating strategies to respond beyond-design-basis events at all units at a site for an indefinite period of time, it is mentioned that it will be inspected “at a later date, after the rule has been finalized”. Do you know at this moment when could the order requirements be implemented in all the plants?	Licensees are being inspected for compliance with the Mitigation Strategies and SFPI Orders, which are being made generically applicable in the rule, as they come into compliance with those orders (as of December 31, 2016, 14 inspections have been completed). Once the rule is in place and rule compliance is required of licensees, oversight will become part of the baseline Reactor Oversight Process. The inspections at that time will be based on the rule, rather than the orders, as is currently the case.	