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Convention on Nuclear Safety

Answers to the Questions

on

Indian National Report

for

7th Review Meeting

Government of India

Convention on Nuclear Safety

Questions Posted To India in 2017

No.	Country	Article	Question	Answer	Support Documents
1.	Australia	General	Whilst not required by the CNS reporting guidelines, suggest that some discussion regarding research reactors may be appropriate, especially the larger ones that could constitute as much risk as a small NPP.	India also agrees with Australia that CNS guidelines do not require discussion on research reactors. India has a few research reactors. However, information of these reactors is not included, as the scope of the convention does not include research reactors.	
2.	Canada	General	The report states, "AERB is continuously augmenting its human resource to meet the demand arising from the expanding nuclear power programme…". (In section 8.1.2.5 the report states that the AERB has 326 staff.) Can the Contracting Party elaborate on how this issue is addressed, and on any difficulties encountered to ensure knowledge transfer to new staff is addressed?	India thanks Canada for posing this good question. Over many years, AERB has been expanding its technical man power through recruitment of personnel at different levels, particularly at the entry levels. The average age of the staff of AERB is currently below 40 and the retirements from the organisation are not many. Section 8.1.2.5 of the National Report describes in detail the status and methods AERB is following with respect to recruitment of staff, their training and knowledge management. AERB has its own training programmes consisting of orientation training for new staff at all levels as well as refresher courses. Further the new staff members are given exposure of the regulatory activities, along with the other experienced staff, for a sufficient period before they are assigned the regulatory responsibilities. This approach has been very effective from the point of view of their knowledge	

				management. AERB also places a lot of importance on documenting the experience related to important safety / regulatory issues, for posterity, in the form of detailed minutes of meetings of safety review committees, review reports, position papers on issues, etc. AERB has also started an intranet based knowledge portal for easy access of all the knowledge resources available with AERB, in an organised manner for reference by its staff. AERB management has places a strong emphasis on maintaining a healthy environment encouraging free discussions on technical, safety and regulatory issues among the staff and for mentoring of the younger staff members by the experienced persons. Apart from its regular staff, AERB also utilises the services of the senior experts, who have retired from regular service, in many of its safety review committees as well as in other assignments as consultants. This has been an added advantage and the younger staff members are encouraged to interact with them on safety and regulatory matters. With such efforts, AERB does not foresee any major challenges in the area of knowledge management.	
3.	Canada	General	The report states implementation of the hydrogen management enhancements is to be completed as part of the long-term actions. Can the Contracting Party explain how the implementation plans were prioritized? Can the	The comprehensive safety assessments carried out for the Indian NPPs post the Fukushima accident and the safety enhancements undertaken in the Indian NPPs were brought out in detail in India's National Reports to the 2nd Extraordinary Meeting (2012) and 6th Review Meeting (2014) of the CNS. As explained in those reports, one of the considerations in the safety assessments was to look at the measures required for	

			Contracting Party provide the proposed completion dates for this work?	ensuring availability of safety functions (shutdown of the reactor, decay heat removal and integrity of the barriers), under extreme situations involving postulated unavailability of all designed sources of cooling water and electric power, apart from the other lessons from the Fukushima accident. The identified safety enhancements based on these assessments were prioritised as short term, medium term and long term actions. The considerations for prioritisation included (a) the safety benefit derived from implementation of the measure, (b) the ease of implementation, and (c) resources required for implementation on the ground. The short term measures were essentially those which could be implemented quickly and consisted mostly of actions to support operator actions under extreme situations for prevention of core damage. The medium term measures included those which involved significant design and procurement efforts as well as detailed planning for implementation on-ground. The long term measures include those which needed significant R&D activities for development and qualification of solutions prior to their detailing and implementation. As of now, implementation of the short term and medium tern enhancements have been completed at all NPPs. Substantial progress has been made in the implementation of long term measures as brought out in Page 24 (Section 6.5.1 of the National Report). Based on the present progress, these measures have been initiated for implementation, to be completed in a phased manner over next two years.	

4.	Canada	General	Can the India National Report be posted on the Atomic Energy Regulatory Body (AERB) website for accessibility by other CANDU operators?	All the National Reports from India for the CNS are publicly available on the websites of AERB as well as CNS page of IAEA.	
5.	Canada	General	Given the status of its NPP program, it is recommended that India become a Contracting Party to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management.	Government of India has not yet decided on the issue of joining the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management.	
6.	Canada	General	Paragraph 3 summarizes inspections of pressure tubes in other Indian reactors, for evidence of the "localized corrosion spots" found in a leaking KAPS-1 tube. It concludes that similar spots have not been observed in other reactors. Please provide details of the number of PHWR reactors that were inspected, and the number of pressure tubes that were inspected in each reactor.	In-situ inspection for detection of localised corrosion spots on the pressure tube exterior surface using BARCIS has been carried out at least in one reactor out of the twin unit PHWR stations. The number of pressure tubes inspected in each PHWR is given below. NPP No. of inspected pressure tubes KAPS-1 - 15 KAPS-2 - 11 NAPS-1 - 4 NAPS-2 - 6 MAPS-1 - 11 RAPS-2 - 4 RAPS-3 - 4 RAPS-5 - 4 RAPS-6 - 4 KGS-2 - 4 KGS-3 - 4	

7.	Canada	General	The 2015 IRRS mission highlighted that the AERB does not have "dedicated competences in the areas of human and organizational factors (HOF)" nor "people who have education and experiences in human factors engineering". This was not addressed in the CNS report. How has AERB addressed this IRRS finding?	Taking account of human and organizational factors is implicit in the regulatory process. The reviewers associated with various activities of regulation necessarily take into account the human and organizational factors. Specific activities which AERB conducts like licensing process of the Management positions of nuclear power plants, licensing process of persons in the shift crew, root cause analysis of the events at NPP take into account predominantly the human and organisational factors. The process of simulator training and re-training in particular takes into account such factors. These are elaborated in Article 11. Further as mentioned in para 13.6 and elaborated in Answer to Q.No.117, due consideration is given to human and organizational factors in all activities from design to operation. In Indian regulatory documents 'Quality Assurance' is a synonym to the requirements for leadership and management for safety. Nevertheless while competencies in the area of human and organizational factors do exist within AERB, AERB has taken note of the observations of the IRRS mission to organise specific number of staff with such competencies in a dedicated group to systematically further enhance these capabilities in the staff of AERB. AERB has taken note of the observation of the IRRS Mission and the suggestion for consideration for ensuring a sufficient number of staff with specialised competence, knowledge, skills and abilities in the area of human and organizational factors (HOF) and communications. To address this suggestion, AERB considered two strategies, (a) to recruit persons with	

				formal qualification in these areas and train them in the aspects related to safety issues of nuclear and radiation facilities and (b) identify the individuals from among the existing technical staff, also having the formal qualification and acumen for these specialisations, and initially pool them into a group depending on the specialisation and provide additional training and opportunities for experiences in human factors engineering. Later a selected additional persons formally qualified on these specialisations can be added. Considering that AERB is primarily a technical organisation in the governmental sector and realising the difficulties involved in identifying personnel having suitable qualifications in both the soft skills as well as in the technical areas; and the issues involved with continued stay of such persons with AERB & their career progression, AERB is pursuing the strategy (b) for addressing the suggestion. With this, AERB will be able to satisfactorily and sustainably address the observation and suggestion of the IRRS Mission within a reasonable timeframe. Further, as brought out in the National Report under article 12 consideration to human factors is being given during full life cycle of the plant by utility as well is covered by variety of means during regulatory oversight. AERB is enhancing the scope of its competence management programme. The enhancement includes the soft kills as well among other multidisciplinary areas, for imparting specialised training / qualifications to the identified personnel. Following this approach, AERB has organised a specialised training programme for all	

				its senior management personnel recently conducted by a consultant on the subject.	
8.	China	General	When talking to the capacity of engineering and manufacturing heavy equipment / components, described in paragraph 1.5. Question: Is there still any gap between the capacity and the industrial need? If so, what is the plan to solve it?	There is no major gap with respect to engineering and manufacturing of heavy equipment for PHWR programme. For indigenous PWRs, development of industrial capacity for major equipment and component is in progress, as briefly brought out in para 1.5 (page 4) of the national report.	
9.	China	General	As India is in the process of setting up Light Water Reactor based NPPs with foreign collaboration in addition to capacity addition with the setting up of new NPPs of indigenous designs. Question: Has the risk and challenge for the design localization been assessed? And how to control them?	Yes, India is in the process of setting up light water based NPPs with foreign collaboration, in addition to capacity addition of indigenous designs. India has a rich experience in the design of indigenous reactors of PHWR technology. In addition to capabilities within the Department of Atomic Energy, India also has a large base of experienced engineering consultancy organisations. These capabilities have been assessed to undertake design localization as considered feasible, and such areas have been identified after assessing risks and challenges. Regulatory review process of the design, construction, commissioning, and operation of LWRs is also well established.	
10.	France	General	The process for a continuous safety level upgrade in NPP under operation is described. It is also mentioned that all these upgrades are taken into account in new built. Could India indicate the safety	The safety objectives applicable for the new reactors are bought out in chapter – 2 of the AERB Code on ‘Design of Light Water Reactor based NPPs’ (AERB/NPP-LWR/SC/D – 2015). The safety objectives as given in the code are brought out below.	

			goals fixed for new reactors in terms of limitation of consequences of severe accidents?	"General Design Objective: To achieve the highest level of safety, measures shall be taken to: (a) prevent accidents with harmful consequences resulting from a loss of control over the reactor core or other sources of radiation, and to mitigate the consequences of any accidents that do occur; (b) ensure that for all the accidents taken into account in the design of the installation, any radiological consequences would be below the acceptable limits and would be kept as low as reasonably achievable; (c) ensure that the likelihood of occurrence of an accident with serious radiological consequences is extremely low and that the radiological consequences of such an accident would be mitigated to the fullest extent practicable; and (d) Incorporate design features such that even in the accident with core melt, only limited countermeasures are needed in the public domain and sufficient time is available to implement these measures. Radiation Protection Objective: The design for safety of a nuclear power plant applies the safety principle that practical measures must be taken to mitigate the consequences of nuclear or radiation incidents on human life and health, and the environment such that event sequences: (a) that could result in high radiation doses or large radioactive releases must be practically eliminated; and (b) with a significant frequency of occurrence must	

				have no or only minor potential radiological consequences. An essential objective is that the necessity for off-site intervention measures to mitigate radiological consequences be limited or even eliminated in technical terms, although such measures might still be required to be taken by the responsible authorities.	
11.	Ireland	General	Ireland thanks India for its comprehensive national report which is structured in accordance with the Convention articles.	India thankfully acknowledges the comment by Ireland.	
12.	Netherlands	General	IRRS-mission: the general statement that India is committed to address the recommendations and referring to the report on the AERB-website does not give information about the recommendations that already have been addressed and how and what concrete actions are in the pipeline. Could you explain when India will plan the follow-up mission.	The actions required with respect to improvements in the regulatory processes as per the recommendations and suggestions of the IRRS Mission to AERB have already been taken. However, effective performance of the regulatory processes with these measures needs to be observed for some time.. Currently AERB is in this phase. Once this phase is complete, AERB will be ready to host the follow up mission.	
13.	Netherlands	General	Could you please explain what you consider to be the most important actions that India will take based on the IAEA Fukushima summary report?	The most important actions from the Indian point view has been to address the lessons learned from Fukushima accident, in the areas of nuclear safety and emergency preparedness and response. Considering these, India has taken swift measures for rigorous	

				safety review of the existing NPPs to address the possible strengthening measures for accident prevention and mitigation. India has also taken action to review and reinforce the emergency preparedness and response framework as well as related exercises and training of personnel. The criteria and guidelines for decision making related to protective actions were also re-looked from the consideration that these actions must do more good than harm. Another area where the preparedness is being enhanced is in the area of strengthening the capacity for implementation of the strategies for dealing with the management of contaminated liquids and solid radioactive material and waste.	
14.	Netherlands	General	The Vienna Declaration also aims at taking measures at existing power plants, if reasonable, to practically eliminate early and large area releases. This recommendation might go beyond the regular PSR's. In the past we might stop by saying "it is not reasonable/impossible to install a core catcher", but with the VD we are challenged to go a step further and pursue improvemens in a more pro-active way. What are the pro-active actions from AERB and the power plants (e.g. by R&D) to further strengthen the nuclear safety in this respect?	As stated in Page 15 of the National Report of India, from the early phase of the nuclear power programme, India has been following a proactive approach towards safety enhancements in the NPPs. Indian regulatory system always placed strong emphasis on learning from experience and using it to enhance safety. This character has helped the nuclear industry, the regulator and the R&D community to evolve with the times to achieve and maintain high level of safety. In line with this, the regulatory system incorporates a system of ‘special safety reviews’ (examples are included on Page 15 of the National Report) undertaken following major events, wherein the implications of such experience and lessons are reviewed for identifying and implementing safety enhancements. Further, as per the existing regulations, the license for	

				operation of NPPs is issued for a maximum period of five years towards the end of which the NPPs may seek a renewal of license. One of the requirements for the renewal is the conduct of a detailed Periodic Safety Review (PSR) at a specified interval, which requires addressing the cumulative effects of ageing and comparison with the current safety requirements / practices, to identify the need for safety enhancements in the existing NPPs [Ref AERB Safety Code on Nuclear Power Plant Operations. Code No. AERB/NPP/SC/O (Rev. 1)]. The details of PSRs practices and experience, including implementation of safety enhancements are detailed in Pages 15, Page 21 (Section 6.3 Periodic Safety Review), Page 22-23 (Section 6.5 Safety Enhancements of Operating NPPs), Page 102-103 (Section 14.1.2.5), Page 156-157 (Section 18.1 Implementation of Defence in Depth). The PSR, as practiced in India, involves identification of shortcomings with respect to the current requirements / practices and identification of the remedial actions / measures. Following these processes, the proactive actions taken for strengthening safety of Indian NPPs include additional water injection points for Heat Transport System, Emergency Core Cooling System, Moderator System, End Shields Cooling System and Calandria Vault Cooling System and spent fuel storage pool have been implemented. Additional air cooled diesel generator, implementation of Containment Filtered Venting System (CFVS), hydrogen management provisions	

				and On-site Emergency Support Center are being implemented to further strengthen accident management. India has a robust R&D programme, which has helped in timely and practically resolving many shortcomings identified as part of PSR, OEF and special safety review processes. Examples of issues already resolved and those in hand are given in Page 24 (Section 6.5.1) and Page 159 (Section 18.2.1) of the Indian National report for the 7th Review Meeting of CNS. India's National Report for the 4th, 5th and 6th Review Meeting also give a number of examples of the back-fits implemented from time to time as considered necessary.	
15.	Pakistan	General	India may like to explain the reasons for omission of RAPS-1 (100 MWe AECL supplied) from safety upgrades, which is shutdown since 2004. Are there any plans for decommissioning of RAPS-1?	RAPS Unit-1 has been under shutdown since October 2004. Presently the reactor core is in defueled state and heavy water is drained from the systems. Prior to the shutdown, all the requirements of safety related systems upgradation were met by implementation of required actions during the shutdown in 2002 for health assessment of coolant channels. Some of the important safety upgrades were reported in the Indian National Report for the 4th Review Meeting of CNS. As of now, the plans for decommissioning of the unit have not been finalised. Presently, all the plant systems are being preserved in accordance with the approved procedures as per a special technical specifications document applicable for the present state of the unit.	

16.	Pakistan	General	India may like to share whether AERB utilizes the concept of Time Limited Aging Analysis (TLAA) for allowing operation of a NPP beyond it's design life.	As per the regulatory practice in India, NPPs are required to undergo PSRs once in ten years. A plant can continue operation; as long as it satisfies the laid down regulatory requirements and demonstrate availability of adequate safety margins. The PSRs involve comparison with current safety requirements and practices as well as assessment of health and ageing aspects of important SSCs. The NPPs are required to develop and implement systematic ageing management programmes, for ensuring health and reliable functioning of the important SSCs. As the plants get older, the ageing aspects receive increasing attention during various safety reviews including PSRs. Methodologies for demonstrating the health of SSCs, in particular the non-replaceable / non-inspectable ones do involve assessment of availability of margins for the specified period.	
17.	Russian Federation	General	According to the IAEA PRIS system, the average capability factor of Indian nuclear units dropped from 89.63 % in 2011 to 76.21 % in 2015. What is the cause of this decrease?	The percentage capacity factors of NPCIL plants in last five financial years i.e 2011-12, 2012-13, 2013-14, 2014-15, 2015-16 were 78.95, 80.16, 83.49, 82.43 and 75.07 respectively. There was a reduction in the overall capacity factor averaged over all the units, in year 2015-16 due to the teething troubles faced in initial operation of KK NPP-1 subsequent to commencement of its commercial operation on 31.12.2014 and long outages of TAPS-1, TAPS-2 & KAPS Unit-2 for varying reasons.	
18.	Russian Federation	General	The Report mentions Prototype Fast Breeder Reactor (PFBR). Could you please give principal characteristics of this facility (fuel	PFBR is a pool-type molten sodium cooled fast reactor, with electric power generation capacity of 500 MWe. It uses a mixed oxide (MOX) fuel of plutonium and uranium.	

			type and enrichment, coolant type)?	A very detailed description of the design of PFBR is given in the National Report of India for the 5th review meeting of CNS (Annexure 18-5, Page 142 to 146).	
19.	Russian Federation	General	What is total electricity generation resulting from the work at the level of the electrical capacity above the installed of all India plants (and what is its per cent from potential nominal generation) in 2013-2015?	The question is not clear. However, assuming that the question is about contribution of nuclear power plants in the overall electricity generation in the country, the following answer is given. The contribution of nuclear power with respect to total installed electricity generation capacity in India in year 2013-14 was 3.5% and for year 2014-15 it was 3.4%. The capacity Factor of NPPs in 2013-14 was 83.49 % [34228 MUs (Rated – 40997 MUs)] and capacity factor for 2014-15 was 82.43 % [35592 MUs (Rated – 43180 MUs)]	
20.	Slovakia	General	It is mentioned that PSA for external events have been developed. Please provide more details about the considered events and hazards and selection criteria?	Methodology has been developed for seismic and external flood PSA. For selection of external hazards, site specific potential hazards are considered, eg. for coastal sites, tsunami, storm surge and precipitation are considered; while for inland sites precipitation and dam failure are considered for external flood analysis.	
21.	Sri Lanka	General	Section 1.1: National Nuclear Power programme indicated that Kudankulam reactors in Tamil Nadu incorporated many advanced passive and active safety features. Can you further clarify what are these advanced passive and active	The design of KKNPP, in addition to the safety features provided in earlier versions of VVER reactors, incorporates additional engineered safety features (ESFs) for catering to design basis accidents (DBAs), Design Extension Conditions (including Severe Accidents), as per regulations and practices adopted in India. For example, the regulatory practice	

			safety features which may not be found in old reactors , and how they help to prevent or reduce consequences of a accident	in India assumes that the off-site power supply may remain unavailable for significant periods and there is further possibility of unavailability of on-site power supply under some conditions. Therefore, the plant needed to incorporate passive and active safety features as part of design, to ensure that the safety functions, including decay heat removal, for extended duration under situations involving unavailability of off-site and onsite power. The plant also have design provisions for ensuring sufficient on-site stock of makeup cooling water and diesel oil for ensuring site autonomy for seven days.	
22.	Switzerland	General	Will India host a follow-up IRRS mission? If yes, when?	Yes. Kindly see the answer to question no 12 posed by Netherland under Article – General.	
23.	Switzerland	General	Were the measures identified in the post-Fukushima reviews undertaken by NPCIL and AERB compiled in a systematic fashion (i.e. into an action plan)? If yes, were they made public?	Yes, systematic compilation was done in the form of reports. The results of safety assessments carried out for Indian NPPs following the Fukushima accident and the action plans for safety enhancements were made public by NPCIL. The details of these assessments along with the outlines of the action plan for implementation of the identified measures/ upgrades were also brought out in the Annual Report of AERB for the year 2011-12 and thereafter progress of implementation were updated in the subsequent Annual Reports. These reports are available publicly on the website of AERB. Further, the Indian National Reports to the 2ndExtraordinary Meeting and 6th Review Meeting of CNS included systematic compilation of the identified safety enhancements as well as the schedule and status	

				of implementation. All these reports have been made public.	
24.	Switzerland	General	In the summary report of the 6th RM, five challenges were identified by the special rapporteur to be adressed by the CPs. Has India taken any measures to respond to these challenges?	Yes. The National Report of India has addressed all the five challenges identified by the special rapporteur. These have been covered adequately in the Summary and under the relevant articles of the report. See below the references to the Sections of the Report where the status on the challenges are brought out: Challenge 1 - Minimising gap between CP's safety improvements: Summary-page11, Sect 6.5, Sect 7.2.1.3, Sect 8.3, Sect 9.5,Sect 9.6, Sect 14.3,Sect 17.3,Sect 18.1 and Sect 19.7. Challenge 2 - Achieving harmonized emergency plans and response measures: Summary- page 12, Sect 16.1, 16.2.7 & 16.7. Challenge 3 - Making better use of operating and regulatory experience and international peer review services: Summary - Page 12 & 13, Sect 6.3 & 6.4, Sect 7.2.1.3, Sect 8.3 & 8.4, Sect 9.5 & 9.6, Sect 11.2.6, Sect 12.3 & 12.4, Article 14, Sect 16.2.6 and Sect 19.7. Challenge 4 - Improving regulator's independence, safety culture, transparency and openness: Introduction - Sect 1.4, Summary-page 13, Sect 7.2.1.1 & 7.2.1.3, Sect 8.2.3, 8.4 & 8.5, Sect 9.4, Sect 10.5, Sect 11.2 and Sect 19.6.	

				Challenge 5 - Engaging all parties to commit and participate in international cooperation: Sect 8.3 & 8.4.	
25.	Switzerland	General	"A. General comments on National Report as a process of self-assessment of the implementation of the obligations of the Convention." The national report covers a lot of information to understand how the CNS-obligations are fulfilled. The national report adresses all aspects of the obligations in Art. 6 to 19 and follows an article-by-article approach. The national report identifies important changes and achievements and highlights significant changes in nuclear safety laws, regulations and practices as well as in safety improvements at existing nuclear installations. The national report reflects compliance with the obligations at the end of every article. The national report adresses international peer review results and include the measures taken to make the results public. The national report makes reference to the IAEA fundamentals and requirements. The national report	The suggestion made by Switzerland is welcome.	

			adresses operating experience and corrective actions to safety significant events. The national report adresses lessons learned from emergency exercises and actions to improve communication with the public witin the summary. The national report includes in Art. 6 a list of backfittings in operating NPP to underline that safety is continuously improved. Reviewing of the national report will be more practicable if the topics highlighted in the summary refers directly to the corresponding articles		
26.	Switzerland	General	"B. Comments on progress made on previous Challenges and Suggestions identified at previous Review." In the summary of the national report most of the suggestions, challenges and planned measures identified at the previos CNS review meeting are expliciteley addressed. The progress in assessing safety culture in the regulatory body is not explicitly mentioned in the summary. However, the regulatory body has initiated a process for assessing the safety culture according to Art. 10. According to the corresponding	AERB recognizes that promotion of safety culture within the NPCIL as well as in the regulatory body is important for securing continual improvement of nuclear safety. AERB's management system identifies safety as a priority and provides for its promotion and continuous improvement. The process for promoting the safety culture includes self-assessment as well as independent assessments. AERB has developed, as part of its management system, a process and internal procedure for assessing its safety culture, using specific questionnaires/ survey. The process was applied initially on a pilot basis, in few Divisions of AERB, resulting in large participation of staff members. The results of the pilot self-assessment were captured in an action plan, implementation of which is in progress.	

			articles most of the challenges are met. Especially the short and mid term Fukushima measures are implemented in all plants. Concerning the long term Fukushima measures (provision for hydrogen management and containment filtered venting) sigificant progress has been made taking into account that both measures have been indigenously developed in India. This challenge is still kept. However, no time schedule for implementing these new systems in the existing NPP is given.	AERB also identified the promotion and oversight of safety culture, both at regulatory body as well as for the utility as one of the Policy Issues in the IRRS peer review mission to India. The aim was to benefit from the global expertise represented by the mission members. For the status / schedule of post Fukushima long-term safety enhancement measures, kindly refer to the answer to question no 3 posed by Canada under Article – General.	
27.	Switzerland	General	"C. Proposals of Good Practices, Challenges, Suggestions." In the summary of the national report one new challenge is identified which refers to safety sigificant events in 2015 and 2016. This challenge is correctly classified as an immediate challenge. Specific Informations about the topics adressed under clause VIII of art. 19 as well as detailed informations about waste management strategy should be provided. In the summary the commitment to implement the IAEA action plan is	Detailed and specific information about waste management strategy is given in Section 19.8 of the National Report. Requirements and guidance on specific aspects related to safe management of radioactive wastes arising during operation of NPPs are specified in the AERB Safety Guidelines no. AERB/NPP/SG/O-11, of which some of the aspects are described in the answers to question no 218 posed by Canada and question no. 219 and 220 posed by Switzerland under Article 19.8. Further, the strategy with respect to management of spent fuel from the Indian NPPs covered in detail in Section 1.3 on 'Nuclear Fuel Cycle', in the Chapter – Introduction. These aspects have been consolidated in the answer to	

			pointed out. However, it is not perceptible in the following articles how this goal will be achieved. The national action plan should be compared with the IAEA action plan to demonstrate compliance.	Question no. 220 posed by Switzerland under Article 19.8 and the same may kindly be referred. Information on how India is fulfilling its commitment to the IAEA Action Plan on Nuclear safety can be seen very clearly in the relevant sections of the National Report. The guidelines for preparation of the national reports under CNS don't give any specific format for presenting this information. The aspects on which India has made specific steps (on peer reviews) since the 6th review meeting of the CNS were therefore brought out in the summary. India has been continuing to fulfil all its commitments on the other elements of the action plan, as stated in the relevant sections of this as well as the national report for the 6th review meeting.	
28.	Switzerland	General	With regard to the implementation of the Vienna Declaration, the national report provides a lot of actions being taken to achieve continous improvement to safety. For example, safety assessments will be performed once in 5 years and once in 10 years, two new important safety codes have been issued in 2014 (site evaluation) and 2015 (design of LWR) and regulatory requirements are reviewed periodically and updated taking into account he latest IAEA requirements.	The comment is thankfully acknowledged.	

29.	Switzerland	General	Principle 1 1.1 How do you define 'a new nuclear power plant'? For example: do you consider a power plant to cease being a 'new nuclear power plant' once operation begins?	The term "new nuclear power plant" is not defined in the Indian regulations. In the National Report for the 7th Review Meeting of CNS, the term 'new NPP' is used in many different contexts. However, for the purpose of the Vienna Declaration on Nuclear safety, India considers 'new NPPs' as those which are given construction consent after the current design code of AERB, ie. Safety Code on Design of Light Water Reactor based NPPs (AERB/NPP-LWR/SC/D) was issued in January 2015.	
30.	Switzerland	General	Prevention 1.2 How does your national requirements and regulations incorporate appropriate technical criteria and standards to address the objective of preventing accidents in the commissioning and operation of new nuclear power plants? For example: can you describe the basic design objectives and the measures you have in place to ensure the robustness and independence of defense in depth measures? Consider for instance inclusion of implementation of Regulatory requirements for: • Robustness of DiD and independency of the levels of DiD;	The national requirements and regulations evolved incorporating technical criteria and standards as appropriate to Indian conditions as well as taking into account the safety standards of IAEA and other international standards. For e.g. the robustness and independence of defense-in-depth measures have been considered since long in Indian PHWRs. Such examples of independence at these DiD levels include maintaining independence between reactor regulating and protection systems, for heat removal use of thermosyphon, use of diesel engine driven pumps as backup for water make-up to decay heat removal systems etc. The regulatory process for establishing as well as revising the safety regulations makes reference to current safety standards of IAEA. This ensures that essential elements for ensuring high level of safety such as DiD, provisions for managing DEC are in line with globally accepted safety norms.	

			• Design Extension Conditions (DEC); • practical elimination of high pressure core melt scenarios; • achieving a very low core melt frequency; • protecting digital safety equipment against Common Cause Failure (CCF). • External events analysis	AERB Safety Code on Design of Light Water Reactor based Nuclear Power Plants (AERB/NPP-LWR/SC/D) addresses the safety objectives and aspects such DiD requirements along with other requirements emanating from the lessons learnt from Fukushima accident. For information on the safety objectives specified in this code for new NPPs, kindly refer to the answer to question no. 10 posted by France to India under the section – General. The code was issued in January 2015. The safety code also requires provision of complementary safety features for mitigating the consequences of severe accidents, should they occur. Further, the design of NPPs shall be such that design extension conditions that could lead to large or early releases of radioactivity are practically eliminated. For design extension conditions that cannot be practically eliminated, only protective measures that are limited in terms of area and time shall be necessary for protection of the public, and sufficient time shall be made available to implement these measures. The design and regulatory assessment of new NPPs will be done to meet these requirements. The aspect of implementation of DiD and the related requirements are discussed in detail in Page 153-157 (Section 18.1.1 and 18.2.2). The information on regulatory reviews for assessing the implementation of DiD are given in Page 99 (Sections 14.1.2.1 through 14.1.2.4.	

31.	Switzerland	General	Mitigation 1.3 How do your national requirements and regulations incorporate appropriate technical criteria and standards to address the objective of mitigating against possible releases of radionuclides causing long-term offsite contamination and avoiding early radioactive releases or radioactive releases large enough to require long-term protective measures and actions. For example: can you describe the measures you have in place to protect against severe accidents and your accident management arrangements - how do you protect staff during accident management? Consider for instance inclusion of implementation of Regulatory requirements for: • Engineered systems to protect the containment; • engineered systems to cool the molten core; • severe accident management, protection of staff during the accident. • Provision and resilience of Emergency Mitigation Equipment (EME)	In addition to the provisions elaborated in the response to Question no 30 posed by Switzerland under Article – General, the AERB safety codes on Site Evaluation of Nuclear Facilities (AERB/SC/S/Rev-1) and Design of LWR based NPP (AERB/NPP-LWR/SC/D) specify the criteria on radiation dose, which shall form the basis of the systems / features for accident prevention and mitigation to be included as part of the NPP design. The dose criteria for normal operation, design basis accidents, and design extension conditions are given in Table – 5 at Page 148 of the National Report. To be able to meet the dose criteria, the NPP design must include engineered systems to protect the containment such as managing containment pressure, reducing containment atmosphere flammability / hydrogen and mitigating large / early releases. For meeting this objective in PHWRs, requirements for maintaining heat sinks within the calandria and the calandria vault are specified. In the context of LWRs, systems to cool molten core, the requirements call for provision of core catcher and water inventory for specified period of core cooling. The specified dose criteria along with AERB guidance document AERB/SG/D-12, while taking into account scenarios specific to PHWR technology seek to ensure that radiation doses to workers at the plant and to members of the public do not exceed the dose limits and that they are kept as low as reasonably achievable in operational states for the entire lifetime of the plant, and that they remain below acceptable limits and as	

				low as reasonably achievable during, and following, accident conditions. The requirements call for the design to ensure that plant states that could lead to large radioactive releases are practically eliminated and that there are no, or only minor, potential radiological consequences for all the plant states with a significant likelihood of occurrence. The aforementioned criteria are to be met by application of DiD and the established engineering principles. The specific requirements with respect to mitigation include provisions for supporting the accident mitigation complementary features for ensuring safety functions during DEC. Further the requirements call for additional provisions for supporting the accident management infrastructure that might be needed to handle extreme events, along with unexpected failure of existing safety features/systems. These aspects are described in the Indian National Report section 17.2.2 (page 148) and section 18.1 (pages 154 – 156).	
32.	Switzerland	General	Principle 2 2.1 How do your national requirements and regulations address the application of the principles and safety objectives of	In addition to the provisions elaborated in the response to Question no 30 posed by Switzerland under Article – General, the AERB safety codes on Site Evaluation of Nuclear Facilities (AERB/SC/S/Rev-1) and Design of LWR based NPP (AERB/NPP-LWR/SC/D) specify the criteria on radiation dose, which shall form the	

			the Vienna Declaration to existing NPPs?	basis of the systems / features for accident prevention and mitigation to be included as part of the NPP design. The dose criteria for normal operation, design basis accidents, and design extension conditions are given in Table – 5 at Page 148 of the National Report. To be able to meet the dose criteria, the NPP design must include engineered systems to protect the containment such as managing containment pressure, reducing containment atmosphere flammability / hydrogen and mitigating large / early releases. For meeting this objective in PHWRs, requirements for maintaining heat sinks within the calandria and the calandria vault are specified. In the context of LWRs, systems to cool molten core, the requirements call for provision of core catcher and water inventory for specified period of core cooling. The specified dose criteria along with AERB guidance document AERB/SG/D-12, while taking into account scenarios specific to PHWR technology seek to ensure that radiation doses to workers at the plant and to members of the public do not exceed the dose limits and that they are kept as low as reasonably achievable in operational states for the entire lifetime of the plant, and that they remain below acceptable limits and as low as reasonably achievable during, and following, accident conditions. The requirements call for the design to ensure that plant states that could lead to large radioactive releases are practically eliminated and that there are	

				no, or only minor, potential radiological consequences for all the plant states with a significant likelihood of occurrence. The aforementioned criteria are to be met by application of DiD and the established engineering principles. The specific requirements with respect to mitigation include provisions for supporting the accident mitigation complementary features for ensuring safety functions during DEC. Further the requirements call for additional provisions for supporting the accident management infrastructure that might be needed to handle extreme events, along with unexpected failure of existing safety features/systems. These aspects are described in the Indian National Report section 17.2.2 (page 148) and section 18.1 (pages 154 – 156).	
33.	Switzerland	General	2.2 Do your national requirements and regulatory framework require the performance of periodic comprehensive and systematic safety assessments of existing NPPs – if so, against what criteria/benchmarks are these assessments completed and how do you ensure the findings of such assessments are implemented?	Yes. For details, kindly refer answer to Question No. 14 posed by Netherlands under Article – Introduction.	
34.	Switzerland	General	2.3 Do your national requirements and regulations require reasonably	Yes. For details, kindly refer answer to Question No. 14 posed by Netherlands under Article – Introduction.	

			practicable/achievable safety improvements to be implemented in a timely manner – if so, against what risk/engineering objective or limit are these judged and can you give practical examples?	Further, a number of practical examples of safety enhancements carried out in Indian NPPs are described in Page 22-23 (Section 6.5) of the National Report.	
35.	Switzerland	General	Principle 3 How do your national requirements and regulations take into account the relevant IAEA Safety Standards throughout the life-time of a Nuclear Power Plant.	India has its own set of national regulations related to all aspects of nuclear power plant lifecycle, which are brought out in AERB Safety Codes and Guides. As stated in Page 15 (Summary Para 5) and Page 33-34 (Section 7.2.1.2), AERB has well-established systems and process for development of regulatory documents which consider in detail the requirements of relevant IAEA documents and feedback from operating experience as well as the national and international current best practices. These regulatory documents are reviewed periodically and updated taking account of the latest IAEA requirements in the relevant area. Further, as mentioned in the Answer to Question No. 14 posed by Netherlands under Article – Introduction; and in a number of Sections in the National Report (Summary, Article 6, 14, 18 and 19), the Indian NPPs are required to undergo regular and systematic Periodic Safety Reviews (PSRs) as a pre-requisite for renewal of license throughout its life time.	
36.	Switzerland	General	General question What issues have you faced or expect to face in applying the Vienna Declaration principles and	‘Acceptable level of safety’ is dynamic which continues to evolve with generation of new knowledge, evolution of safer technologies and expectations of the public. Accordingly, continuous safety up-gradation has been integral to the safety	

			objectives to your existing fleet or new build of Nuclear Power Plants	assessment process mandated by AERB for existing as well as new builds. As per regulatory requirements, the license renewal for existing NPPs is subject to the regulatory acceptance of the outcome of the assessment against the current safety requirements / practices. While revising the regulatory documents, relevant IAEA safety standards are referred among other sources. These inherent attributes to the regulation have facilitated the application of principles of Vienna Declaration in safety regulation of existing NPPs as well as upcoming projects without any specific issues.	
37.	Ukraine	General	Are NPPs in India (which is a densely-populated country) going to perform level 3 PSA? Are there requirements of the regulatory body for the performance of level 3 PSA?	India has carried out limited Level-3 PSA (for specific accident sequences emanating from an identified NPP) to demonstrate the capability of performing the full scope PSA. However, it is to be noted that the numerical safety targets for surrogate measures of risk (i.e. core damage frequency and large early release frequency) derived from Level-1 and Level-2 PSA are set such that they are commensurate with limiting the public risk. As per AERB regulations, internal event plant-specific Level 1 PSA (full power) is mandatory for all NPPs. For new NPPs, Level 1 PSA (full power) needs to be completed prior to first criticality and for NPPs in operation, it shall be updated and presented as a part of periodic safety review (PSR), which is conducted every 10 years. Recently, the scope of the PSA has been increased to include all modes of operation including shut down. AERB safety code contemplates	

				to increase the scope and levels of PSA to include external events and Level-2 PSA. Performing Level-3 PSA is also recommended to assess the adequacy of emergency preparedness and response plans.	
38.	Ukraine	General	This section indicates that the nuclear facilities in India were sited, designed, constructed and commissioned and are operated in accordance with strict quality and safety standards: why the list of life cycle stages does not include decommissioning of nuclear facilities?	The regulatory framework in India cover all stages of NPP lifecycle, including siting, design, construction, commissioning as well as decommissioning. The regulatory requirements and regulatory processes for all these stages are well established, as described under the relevant articles in the national report. None of the nuclear power plants (NPPs) in India have been decommissioned so far. The compliance to quality and safety standards will be ensured when decommissioning of NPPs is taken up .	
39.	Ukraine	General	Is regeneration of uranium and plutonium used in the processing of irradiated nuclear fuel? How the target products resulting from spent nuclear fuel processing (U, Pu, Np isotopes) are further managed?	The reprocessed Uranium and Plutonium from NPPs will be used for India's second stage nuclear power program.	
40.	Ukraine	General	Have requirements been established for risk-informed decision-making? If yes, what quantitative criteria for their application have been identified? What upgrades or administrative and technical measures have been implemented and/or planned for	(i) The risk component is included in the decision making. Quantitative criteria for risk informed decision making are incremental CDF and incremental conditional core damage probability. (ii) As part of accident management, preventive and mitigating measures have been implemented. For PHWRs, accident management guidelines are	» Note on Features of Indian PHWRs

			the ex-vessel phase of severe accidents? Is it planned to enhance qualification requirements for the design equipment involved in mitigation of severe accidents? Does the severe accident management guideline include ranking of personnel actions in case of a severe accident at multiple units at the same time? If yes, how the technical and human resources are redistributed? Does the methodology for determining human errors in PSA take into account additional stress caused by increase in peer reviews (internal and by external organizations)?	designed to have in-vessel retention, for which sufficient time is available, owing to comparatively slower progression of the accident (Please refer Attachment titled "Note on Features of Indian PHWRs"). VVERs are provided with core catcher. (iii) At present First Generation HRA methods such as Technique for human error rate prediction (THERP), Human Cognitive Reliability (HCR), accident sequence evaluation program (ASEP), etc. are used for estimation of human error probabilities. Maximum stress levels as per these are considered. No special emphasis for additional peer reviews is mentioned in these methods.	
41.	United States of America	General	The utility performed safety assessments for TAPS-1&2, KAPS-1&2 and MAPS-1&2 as part of the PSR. Based on the satisfactory review of the report of these assessments, AERB renewed the licenses for operation of these NPPs. Can you share some of the findings and lessons learned from these reviews?	Since the 6th review meeting of CNS in 2014, PSR were performed at TAPS-1&2, KAPS-1&2 and MAPS-1&2. These PSRs involved review of the identified safety factors for these NPPs in comparison with the current safety requirements and practices as well as assessment of operating experience and cumulative effects of ageing, to identify the need for safety enhancements. The safety analyses of these NPPs were also reviewed against the current requirements on PIEs, analytical methods / models, assumptions and criteria to identify the need for revisions. The current PSR was the second such	

				comprehensive safety review carried out for each of these NPPs. TAPS 1&2 and MAPS 1&2, which belonged to the older generation NPPs underwent their first round of reviews between 2000 and 2006, based on which significant safety enhancements were implemented at these units as well as their safety analyses were revised. The details of the safety assessment of these NPPs and the safety enhancements implemented were reported in the Indian National Report for the 4th review meeting of CNS which was held in 2008. In the present PSR, the issues concernedincluded mainly of the safety enhancements identified and being pursuant to the post Fukushima safety review of the Indian NPPs. The reviews have also shown that with the systematic programmes for aging management and equipment qualification, instituted following the earlier round of reviews, the ageing aspects of important SSCs, obsolescence management and maintenance of equipment qualification were adequately taken care of. Significant amount of work related to health assessment of the Reactor pressure Vessels (RPV) of TAPS 1&2 reactors were carried out, which included inspection of weld joints, involving enormous amount of work for developing the inspection systems and assessment methodologies. The reviews have shown that the safety performance of the NPPs have remained satisfactory. Significant progress has been made in the implementation of the identified safety enhancements and schedules have been finalised for implementation of the measures in	

				progress. Based on the satisfactory results of the assessments, renewal of operating licenses for these NPPs were agreed.	
42.	Australia	Article 6	The introduction to section 6.2 states that there were 111 significant incidents during the period 2013-2015 and that only 2 were rated at INES Level 1. Section 6.2.1 then identifies an additional incident from 2016 that was also rated at INES Level 1 whilst the INES rating for the RAPS-2 incident (section 6.2.2, also in 2016) is not provided. Please identify how many other significant incidents occurred over the whole period covered by this report (i.e. including those from 2016) as the proportion of INES Level 1 events to the total number of events can be a useful indication of overall safety.	During the reporting period i.e. from 2013 to August 2016, a total 131 events were reported from operating NPPs. Out of these, 2 events were rated at level 1, while one event (i.e. KAPS-1 pressure tube failure) was assigned provisional rating of level 1. RAPS-2 incident of leak from primary coolant system on January 29, 2016 was rated at level 0 on INES.	
43.	Australia	Article 6	Has the elevation of the SBO DG and its associated cooling towers had an impact on the seismic qualification of these components?	Yes. Subsequent to raising of elevation of SBO DG and its associated cooling tower, these have undergone seismic re-evaluation.	
44.	Canada	Article 6	The significant event described in this section regarding the release of	The event was reviewed at all stations. The following design / procedural modifications as per their	

			tritiated contaminated water resulted in procedural changes as stated in the report. Can the Contracting Party comment on whether or not the licensee considered the use of design changes to the dyke and drain systems to prevent the potential for a weather event to allow tritiated contaminated water to escape to the environment?	applicability have been implemented at all stations: 1) Most of the drains pipes with valves and blind flanges directly communicating dyke area to storm drains have been either deleted & sealed or the drain pipes are plugged and valves were chain locked in close position. These barriers (valve & blind flanges / plugs) are covered under preventive maintenance and periodic surveillance programme to avoid recurrence of such events. 2) Sump Transfer pumps along with sampling provision have been installed in the dyke area sumps. The provisions have been made for transfer of dyke area water either to downgrade heavy water storage tanks or liquid waste storage tanks or storm drain based on sample results. Spectacle blind flange along with the valves have been provided in the transfer line going to storm drains. 3) Floor beetles have been installed in the dyke area to alert operator for any water ingress / spill in the area to initiate early corrective action. The implementation of the above changes will prevent the potential for a weather event to allow tritiated contaminated water to escape to the environment.	
45.	Canada	Article 6	In the case of the KAPS-1&2 coolant channel leaks described in section 6.2.1 (pp.19-20) it appears that OPEX from these events has not been disseminated with the international community. Can the Contracting Party explain	The KAPS-2 & KAPS-1 events and the information on the investigation findings were shared with the international nuclear community through the following. • Annual Meeting of the Senior Regulators from the Countries Operating CANDU Type reactors in November 2015. • Event Rating Form for KAPS-1 event posted on	» Note on KAPS PT Failure

			if they considered providing OPEX to other NPPs with pressure tubes? Please elaborate on the difficulties that have delayed sharing the safety significant operating experience from these two events through the existing mechanisms (such as the IAEA INES reporting system, the CANDU Owners Group OPEX meetings and WANO Tokyo office)? Can the Contracting Party provide their plans to disseminate detailed OPEX information on the KAPS 1&2 pressure tube leak events to the international nuclear community (particularly CANDU licensees)? Can the Contracting Party respond to the suggestion that the upcoming COG Fuel Channel Seminar (May 2017) presents an opportunity to share details of these events. Challenge: Improve the timeliness and extent of sharing of safety significant information with international bodies, other operating organizations and	IAEA-INES website on March 14, 2016 • AERB Press Releases, after KAPS-1 event, on March 11, 2016, March 14, 2016, March 16, 2016, March 22, 2016 and July 1, 2016. These are still available on AERB website. • Communications with CNSC, Canada following KAPS-1 event • Bilateral Meeting with Canadian Delegates on the side-lines of the IAEA International Conference on Effective Nuclear Regulatory Systems during April 11 – 15, 2016 at IAEA Headquarters, Vienna • IRS report on KAPS-2&1 events posted on IAEA-IRS website on October 14, 2016. • Technical Meeting to exchange experience on recent events in NPPs and Meeting of Technical Committee of IRS National Coordinators during October 17-20, 2016 at IAEA Headquarters, Vienna • Biennial Meeting of INES National Officers during November 21-25, 2016 at IAEA Headquarters, Vienna • Bilateral Meeting with CNSC Officials on the side-lines of the IAEA General Conference in September 2016 at IAEA Headquarters, Vienna • OECD/NEA WGOE presented the KAPS events in Committee on Nuclear Regulatory Activities (CNRA) & Committee on the Safety of Nuclear Installation (CSNI) meetings in November & Dec 2016 respectively. Queries raised were answered by Indian representative. • Annual Meeting of the Senior Regulators from the Countries Operating CANDU Type reactors in February 2017. The events occurred at KAPS-2& KAPS-1 are first of	

			regulatory bodies through existing mechanisms.	a kind. The investigations to find the root cause of the events are still in progress. Once the root cause is identified the relevant information will be shared with the nuclear community. For further details on the event and an update on the progress of investigations, kindly refer attachment titled 'Note on KAPS PT Failure'. In view of the information shared with the international nuclear community progressively, as indicated above, India does not consider sharing of safety significant information as a challenge.	
46.	Canada	Article 6	Paragraph 4 notes that in light of the pressure tube leakage events in KAPS-1&2, NPCIL "augmented" its pressure tube inspection program. This is commendable. Please explain how the inspection program has been augmented (for example, by increasing the number of channels examined in each inspection?) and whether the augmented program is being implemented at all PHWR reactors.	The in-service inspection program of coolant channels for all PHWRs has been modified to include periodic inspection for detecting the localised corrosion on the exterior surface of pressure tubes.	
47.	Canada	Article 6	Paragraph 5 describes NPCIL efforts to investigate the leaking pressure tube in KAPS-2. The authors suspect that "The failure mechanism… is similar to failures typical of CANDU experience."	The events occurred at KAPS-2 & KAPS-1 are first of a kind. The investigations to find the root cause of the events are still in progress. For further details on the event and an update on the progress of investigations, kindly refer attachment titled 'Note on KAPS PT Failure'.	» Note on KAPS PT Failure

			We assume the authors are referring to Delayed Hydride Cracking (DHC), the only mechanism observed to cause through-wall cracking of Canadian pressure tubes. DHC cracks can initiate for a number of reasons; has NPCIL identified the root-cause in the present case?		
48.	Canada	Article 6	Paragraph 6 mentions the "…prolonged exposure to steam [originating with the cracked PT]… could have led to the localized corrosion." Since paragraph 1 suggests that KAPS-2 operators quickly reacted to increasing Annulus Gas System moisture levels (by shutting down the reactor), it is not clear what is meant by "prolonged". Please elaborate.	Initially it was "suspected" that the localized corrosion of pressure tube exterior surface are secondary effect of leaking coolant and might have occurred due to 'suspected' prolonged exposure to steam environment following leak from pressure tube. The extensive literature survey done after the KAPS-2 event also indicated that the time period required for such corrosion to form on Zr-2.5 Nb alloys is considerably long. Thus it was suspected that a minor leak in the pressure tube might have been present for a prolonged period and AGMS was not sensitive enough to indicate such a minor leak. However a thorough review of the past records of AGMS indicated that the system was well maintained and responding. This review did not indicate that the pressure tube was leaking for long time. The performance evaluation of AGMS at other PHWR (similar to KAPS-2) also confirmed that the system is sufficiently sensitive and even detects a leak much lower than the system design basis. For further details on the event and an update on the	» Note on KAPS PT Failure

				progress of investigations, kindly refer attachment titled 'Note on KAPS PT Failure'	
49.	Canada	Article 6	Inadvertent radiation exposure of radiation worker at TAPS-3&4 on May 17, 2014. a What are the corrective actions implemented to avoid a reoccurrence of this event? b Can India provide some insights and/or details on the lessons learned? c Has there been a follow-up on this event after the review by AERB?	Even though the radiation exposure received by the worker was well below the regulatory dose limit for occupational worker, the event was taken seriously as it indicated need for strengthening the work procedures related to handling of radioactive materials. The event was investigated to establish the root cause and contributors; and outcome of the same was reviewed within the utility and at AERB. Based on the reviews, procedures and administrative controls related to handling of radioactive material were relooked at. Accordingly, enhancements were carried out in the procedures and administrative controls related to transfer of irradiated neutron detectors with aim of reducing the potential for human errors. Appropriate augmentation of automated alarm system was also carried out and contingency plan were developed for this specific activity. The important lessons learnt from the event are as follows : i. The event highlighted the importance of effective implementation of error reduction tools such as “Self-check” and “Peer check”. ii. The work procedure should also consider occurrence of possibility of two independent failures and ensure successful implementation of error reduction tools. The follow up measures taken based on this event were indicated in the Indian National Report section	

				6.2.4, which included suspension of activities related to handling of irradiated neutron detector. Further transfer of irradiated neutron detectors was permitted by AERB only after verification of satisfactory implementation of the necessary corrective actions by the utility to prevent occurrence of such event in future. The operating experience and lessons learned related to this event were widely shared among other Indian NPPs. Based on the review, refresher training was imparted to plant personnel on human error reduction tools. The operating experience and lessons learned from the event were also shared internationally (Incident Reporting System Report: IRS/8423).	
50.	Canada	Article 6	Paragraph 2 explains that NPCIL employs the BARCIS tool to perform in-service inspections of pressure tubes. Based on Article 6.2.1, we surmise that the tool provides volumetric and dimensional information about each tube. Article 6.1.4 indicates that NPCIL also periodically monitors "hydrogen content". Are such measurements also made using the BARCIS tool?	A separate slivering tool (other than BARCIS) is deployed for collecting material samples from pressure tubes for hydrogen analysis.	
51.	China	Article 6	As an inadvertent release of tritium activity to storm water drain occurred at NAPS in June 2013, partly because of the absence of corresponding procedure.	The operating experience gained from the event of escape of tritium activity to environment through dyke area at NAPS was disseminated to other NPPs. The event was reviewed by all NPPs and based on the outcome of the review, measures such as modification	

			Question:What has been or will be done to avoid the incompletion of procedure in other NPPs?	in design and procedures were implemented in all NPPs as per their applicability. Also kindly see the answer to question no. 44, posed by Canada under Article -6.	
52.	China	Article 6	As an inadvertent radiation exposure of radiation worker occurred at TAPS-3&4 on May 17, 2014, partly because of the ineffective implementation of human error prevention tools like self-check, peer check, supervision. Question: What has been or will be done to avoid this kind of human error in other NPPs?	Kindly see answer to Question : 49, posed by Canada, under Article 6.	
53.	China	Article 6	As both KAPS-1 and KAPS-2 encountered an incident of coolant leakage, in 2016 and 2015 respectively. Question: What has been or will be done to avoid the occurrence of similar incident in other 8 NPPs of the same design?	The events occurred at KAPS-2 & KAPS-1 are first of a kind. The investigations to find the root cause of the events are still in progress. For further details on the event and an update on the progress of investigations, kindly refer attachment. Based on the insights gained so far from the investigation findings, following corrective measures have been taken. • The specifications as well as quality checks of the gases used in AGMS have been strengthened in all PHWRs. • The pressure tube exterior surface of the coolant channels in other operating PHWRs have been inspected and observed to have no localized corrosion.	» Note on KAPS PT Failure

				• The inspection for detection of localised corrosion has been included in the ISI program of coolant channels.	
54.	Netherlands	Article 6	Things like self-check, peer check and supervision: are these tools required to apply in the AERB regulations? The application of those are part of a robust safety culture: has AERB required NPCIL to do an evaluation of the application of these tools in their plants? Has AERB included this in their inspection programme?	The self-check, peer check and supervision tools are a part of procedures developed by the utility. As per the AERB Safety Code “Quality Assurance in Nuclear Power Plants (NO. AERB/NPP/SC/QA), utility has to carry out an independent assessments to measure the adequacy of work performance to monitor item and service quality and to promote improvement. . During regulatory inspection, the implementation of these tools are checked on sample basis, as a part of compliance checks to the procedures developed by the utility.	
55.	Netherlands	Article 6	Operating experience programme is explained in the Indian report. For foreign incidents it seems that IRS is used as the only source of input in the proces, but these contain mainly incidents from INES level 2. Since also incidents below that level can be of interest, can India explain how that information will be gathered? It also would be interesting to know how India is processing so-called regulatory experience feedback (REF).	The AERB Operating Experience program is detailed in the Section 19.7 of CNS Report. This program also utilizes international operating and regulatory experience gained from IAEA-IRS, IAEA-INES, international peer review reports (such as CNS, IRRS), Bi-lateral & multi-lateral co-operations with other regulatory agencies and regulator’s forums. Any experience, irrespective of the INES rating, which is considered useful for international nuclear community is exchanged through IAEA-IRS. The utility and the NPPs have their own programme for OEF, which involves collection and review of reports international events through IAEA-IRS, WANO, COG, etc. for learning lessons. The AERB also utilises the regulatory experience gained from national regulatory processes (like licensing, inspections, safety review & assessment)	

				and bi-lateral & multi-lateral co-operations with other regulatory agencies & regulator's forums. The inputs are screened and review & analysed in AERB for development of actions for improving the safety of NPPs and regulations.	
56.	Netherlands	Article 6	PSR: normally apart from current (modern) regulations, also the existing plants are compared with newer designs that have been introduced. Is this also part of the Indian PSR? Can India elaborate on the trending of incidents, the documenting in database, rootcause analysis, using of precursors and how the roles of the licensee and AERB are in the OPEX proces?	Yes, it is part of PSR. As mentioned in the Summary of the National Report, page 15, India has been following an active nuclear power programme, with units being added more or less at a regular pace. With India pursuing an indigenous nuclear power programme, the NPP designs have been seeing enhancements over time, particularly in respect of safety, in tune with the prevailing international benchmarks and best practices. This has facilitated the design approach for the Indian NPPs to stay up to date with the state of art. During the PSR the safety factors for the NPP are assessed in comparison with the current requirements / practices, a practical approach of which includes comparison with the latest design plant of similar type. India has a robust OPEX programme. The features of the OPX programme are discussed in detail in section 19.7 of the National Report, on "Operating Experience Feedback Programme, wherin the scope of the programme, role of different agencies, the processes etc. are detailed. The OPEX system includes systems for reporting of events, screening, investigations and analysis,	

				corrective actions development and management programme, trending and review process, utilisation of OE and dissemination of OE information, monitoring of OE programme, monitoring of OE programme effectiveness and Quality Assurance. Both AERB and the licensee maintain separate databases relating the records of various aspect of the programme. The trends as reflected by the elements of OE program are periodically reviewed to identify any generic concerns and to initiate changes in the OE as well as regulatory activities. The precursors to significant event are also identified as a part of event reporting system and corrective actions to prevent recurrence of the event are taken. Root cause analysis is done in accordance with standard practices and various analysis methods are applied. Further details are availble in AERB safety guide AERB/SG/O-13 "Operational Safety Experience Feedback On Nuclear Power Plants", which is available in AERB website.	
57.	Netherlands	Article 6	The national report states that India closely follows the IAEA regulations and has a extensive OPEX programme. However, the implementation of PARs, Filtered Venting and SAMGs started after Fukushima. Could you present your view on this?	The provision for handling severe accident were under development even before the Fukushima accident (ref CNS report of India for the 5th Review Meeting prepared in August 2010). This was also a regulatory requirement as per AERB Safety Code AERB/NPP-PHWR/SC/D on 'Design of PHWR based NPPs' published in 2009. The Fukushima accident further prompted for expeditious development, enhancement and implementation of SAMG provisions.	

				Kinndly also see the answer to question no. 129 posed by Canada under Article - 14.	
58.	Netherlands	Article 6	What are the INES levels determined for the events reported?	The event of leakage from the weld joint in the feeder pipe at RAPS-2 described in section 6.2.2 was rated at level – 0 in the INES. Please refer answer to question no. 42 posed by Australia under Article - 6.	
59.	Russian Federation	Article 6	The Report provides information about operating Indian NPPs. Could you please give information about operating research power nuclear installations.	India has a few research reactors. However, information of these reactors is not included, as the scope of the Convention does not include research reactors. Please also refer to answer to Question no. 1 posed by Australia under Article - General.	
60.	Slovenia	Article 6	The radiological impact due to operation of NPPs on the environment for each site is monitored by the Environmental Survey Laboratory (ESL) , which is established by BARC (a TSO 18 of AERB) well before the commencement of operation of NPP. Q.: Does the laboratory have accreditation according to international standards and has its own quality management system?	Yes, ESLs are accredited consistent with International standards. All ESLs participate in the International inter comparison exercise of IAEA. ESLs have own quality management system. These ESLs are ISO certified for integrated management system in EMS (Environment Management System, OHSAS (occupational health and safety management system) and QMS (Quality Management System).	
61.	Slovenia	Article 6	Further transfer of irradiated neutron detectors was permitted	Please refer to the answer for Question 49 posed by Canada under Article - 6.	

			only after satisfactory implementation of the necessary corrective actions by the plant to prevent occurrence of such event in future. Q.: What kind of corrective action has been taken?		
62.	Switzerland	Article 6	To what extent the current low collective dose for KKNPP-1 of 0.1 person Sievert is due to the short time since the beginning of operation? What collective dose is expected for KKNPP-1 in the future?	KKNPP-1 reactor attained first criticality inJuly-2013, reached 1000 MWe power level in June 2014. First re-fueling shutdown was taken during June 2015 – Jan 2016. The current low collective dose for KKNPP-1 of 0.1 Person-Sievert can be attributed to this initial period of operation. As per the world average, the annual collective dose per unit for PWR reactors is about 0.5 Person-Sievert. It is expected that the collective dose in the long term for KKNPP would be of a similar order.	
63.	Switzerland	Article 6	Why did the planning of containment filtered venting systems (CFVS) start only after the Fukushima event and not when the first CFVS were implemented in other countries?	Indian PHWRs, as part of design have certain inherent characteristics / features available, as explained in Attachment titled “Note on Features of Indian PHWRs”. Analysis for beyond design basis accident scenario (design extension conditions) and development of accident management guidelines were in progress at the time of Fukushima accident, and based on the accident analysis, requirement of containment venting was envisaged (in select PHWR units). Like in the rest of the world, these activities were accelerated post	» Note on Features of Indian PHWRs

				Fukushima accident, and containment filtered venting system was developed.	
64.	Switzerland	Article 6	What are the advantages of the indigenously developed CFVS compared to already existing CFVS in other countries?	Development of indigenous technology has its own spin-off benefits; all information related to the complete R&D, system design basis and details are available with the utility, which will help in maintaining and improving the system.	
65.	Ukraine	Article 6	Do the post-Fukushima safety upgrading measures include the development of conceptual decisions on management of large volumes of radioactive water generated during mitigation of beyond design basis accidents? If yes, what is the implementation status and basic provisions of these conceptual decisions?	A conceptual scheme for handling large volumes of radioactive liquid waste generated during beyond design basis accident scenario with various details including estimation of volume, activity level, removal/transfer of waste, treatment and disposal of treated waste has been prepared. The large volumes of radioactive liquid wastes can be stored in the available space inside Reactor Building (RB). This feature allows retaining radioactive water inside RB for longer duration till the time activity to be handled gets reduced to very low level. Provision for transferring of such liquid waste from RB to outside for treatment and disposal has been made. The scheme/arrangements are of the type that can be made available when required, since immediate treatment and disposal is not envisaged, as stated above.	
66.	United Kingdom	Article 6	The National Report describes post-Fukushima safety enhancements and mentions pressurised heavy water reactor (PHWR) and boiling water reactor (BWRs) stations. Please explain	As stated in page 2 of the Indian National Report for the 7th Review Meeting of CNS, the KK NPP 1&2 reactors incorporate many advanced passive and active safety features. Post Fukushima, extensive safety review of all Indian NPPs, especially with respect to external events was undertaken and the	

			whether any safety enhancements have been made to the pressurised water reactors at the Kudankulam Nuclear Power Plant (KKNPP).	findings were presented in the National Report for the 2nd Extraordinary Meeting of CNS. The original design of KKNPP itself had sufficient features to address Fukushima like accident conditions, including passive safety features. Based on post Fukushima safety review of KK NPP 1&2, which were under construction / commissioning at that time, a number of safety enhancements were implemented, as part of further enhancement of safety over and above the originally designed systems/features for handling extreme external events.	
67.	United Kingdom	Article 6	The report states that "All the nuclear power plants have established the in-Service Inspection (ISI) programme approved by Atomic Energy Regulatory Board (AERB)". However, the report does not provide details of the codes and standards utilised in establishing the in service inspection programmes concerned with ageing management of pressure retaining structures and components. Please provide details of the following in relation to in-service inspection programmes for all safety related systems which constitute a pressure boundary:	In-Service Inspection programme is the subset of ageing management programme. In-Service Inspection Manual for each plant is developed, which encompasses all the SSCs (Systems, Structures and components) including components / equipment in primary pressure boundary. The codes and standards utilized in establishing the In-Service Inspection program are: i. AERB Safety Guide AERB/SG/O-2 , In-service Inspections of NPPs ii. IAEA Safety Guide No NS-G-2.6- Maintenance, Surveillance and In-service Inspection in NPPs. iii. ASME B & PV Code , Section XI iv. PNAEG 07-008-89 Code for KK NPP v. CAN/CSA-N 285.4-14- Periodic Inspection of CANDU NPPs components. In addition, guidelines are issued by the utility with respect to FAC on Secondary System Piping and	

			• Choice of codes and standards utilised in establishing the programmes, • How ageing management has been incorporated in the programmes	components. The verification of ageing management and ISI programme is done during regular reviews and as part of the comprehensive review during the PSR.	
68.	United States of America	Article 6	The report states that KKNPP-2 achieved criticality on July 10, 2016, and the unit is in advanced stage of commissioning for power operation. Please provide an update on the status of the plant.	KKNPP unit 2 after having achieved criticality, was synchronized to the southern grid of India for the first time in August 2016. As per the regulatory requirements, the unit is currently undergoing phase – C commissioning, involving high power operation in stages of 50%, 75%, 90% and up to full power, for completing the balance of phase C commissioning tests.	
69.	United States of America	Article 6	Can you provide an update on the results from the root cause analysis and investigation of pressure tube leaks at KAPS?	The events occurred at KAPS-2 & KAPS-1 are first of a kind. The investigations to find the root cause of the events are still in progress. For further details on the event and update on the progress of investigations, kindly refer the attachment titled “Note on KAPS PT Failure”.	» Note on KAPS PT Failure
70.	Australia	Article 7	Suggest that for future reports, this section (and possibly other sections) could be simplified by referencing previous reports and only identifying changes since the last report.	The comment is acknowledged.	
71.	Canada	Article 7	The report states that “in certain cases AERB may opt for alternative review process as	The alternative review process may be one or two tier review process instead of three tier. This is as per graded approach	

			deemed necessary". What would the "alternative review process" include? What are the factors taken into account when opting for the alternative review process?		
72.	Switzerland	Article 7	"The 'Nuclear Safety Regulatory Authority (NSRA) Bill 2011', which expired, aimed at establishing the regulatory body under the new legislation. A similar bill is being processed." What will the significant changes be and how will the new bill strengthen the legal framework for safety regulation of safety in nuclear facilities as well as radiation facilites and associated activities?	The Nuclear Safety Regulatory Authority (NSRA) Bill, 2011 was introduced in Parliament to enhance the existing 'de facto' independent status of AERB to 'de jure' independence.	
73.	Netherlands	Article 7.2.1	A new proposal for a NSRA: what weaknesses in the current structure have to be solved? What are the proposed changes to the current situation?	Please refer answer to Question No. 71 posed by Switzerland under Article - 7.	
74.	Netherlands	Article 7.2.1	Does India have embedded in its regulations the concept of continuous improvement of safety? If yes could India elaborate on this? For the development of regulations AERB seems to lean very much on the IAEA standards.	The concept of continuous improvement of safety is embedded in the practices and regulations. The requirement and practice of PSRs, special safety reviews of NPPs and timely implementation of the identified safety enhancements in the NPPs are evidence of this. Section 6.3 and 6.5 of the National report of India describes these aspects in detail.	

			Are there also other sources of regulation which are used? E.g. Wenra reference levels or Wenra Safety objectives for new reactors?	Considering the importance of these aspects, they are also included in the summary of the National Report. The regulatory documents of AERB are updated periodically based on experience and scientific developments taking into account recommendations of IAEA safety documents as brought out in the summary and section 7.2.1.3 of the National Report. AERB has carried out a comprehensive review of the prevailing safety requirements to ascertain and to identify the need for revision in the requirements and guidance documents, in light of lessons learnt from Fukushima accident. The revision of these documents is being done in a progressive manner. While preparing / revising the regulatory documents, the requirements / guidance available in other relevant international regulations, primarily that of the IAEA Standards as well as other relevant international standards / practices are also suitably considered, with the intent of adopting the best practices. for example, the recently, published AERB safety code on 'Design of light water based NPPs' (AERB/NPP-LWR/SC/D, 2015) has used other sources of regulations such as WENRA apart from IAEA references.	
75.	Germany	Article 7.2.2	Regarding the Indian system of licensing, could you please describe if and how the public and interested parties are involved during the licensing process? Also we would like to know what legal provisions India has to prevent the	During the stage of environmental clearance for siting, the Ministry of Environment, Forests and Climate Change public hearings are conducted. For conduct of safety reviews, AERB has a committee system, wherein provisions are made for obtaining the stakeholders views, including from the utility. Similar system exists for drafting of the regulations. AERB has now instituted a practice of obtaining comments	

			operation of a nuclear installation without a valid license.	from the public on its new / revised draft regulatory requirement documents, before their publication. The public and the stakeholders can also comment on the existing documents, which would be considered whenever the document undergoes subsequent revision. The key committees of AERB have membership from various academic institutions, other Government Departments, apart from the nuclear safety experts from AERB and the TSO. AERB shares detailed information regarding the issuance of consents and the related review / assessment findings to the public promptly through press releases and through its annual reports, which are posted on the website of AERB. The current laws in the country, the Atomic Energy Act, 1962 and the Atomic Energy (Radiation Protection) Rules, 2004 (in the earlier version the Radiation Protection Rules, 1971) prohibit the establishment and operation of nuclear installations without a valid license from the Competent Authority.	
76.	Russian Federation	Article 7.2.2	Is there any difference in approaches to reviewing licensing documentation of different NPPs (of small and large power; with different reactor technologies)? If there is no difference, then from existing licensing experience, what	The legal requirements as well as approach with respect to licensing of NPPs and review of licensing documentation for all types of reactors are essentially the same. However, some enhancements with respect to scope and detailing of the reviews can be expected depending on the use of specific standards used in design / construction, use of first of a kind systems,	

			are advantages of common approach to reviewing NPPs with different technologies and capacity?	etc. This aspect is brought out in Articles 14 & 18 of the report. The common approach helps in evaluation against uniform safety objectives and criteria. The legal requirement / basis also are applied uniformly. This helps to avoid inconsistency in the regulatory reviews.	
77.	Netherlands	Article 8	Given the list of AERB participation in WG of NEA CSNI/CNRA it appears that there is no participation in the WGHOF. This seems to be consistent with the last two questions. There's also no participation in the special group on safety culture and the working groups on eletrical systems (WGES) and External events (WGEV). Please explain.	Kindly note that India is not a member of OECD-NEA and participation of India in various working groups is by invitation. With the expanding nuclear programme of India, the resources needed for the safety reviews are given more priority. For optimal utilisations of resources, an approach involving participation in the selected forums working groups is being followed currently.	
78.	Switzerland	Article 8	“AERB is currently augmenting its staff strength to reach about 450 in the near term.” Question: What is the reason to expand staff strength from currently 326 to about 450?	AERB has been working on the enhancement of human resource base for some time, with the objective of catering to the regulatory review / monitoring requirements of the expanding nuclear programme as well as the enlarging base of radiation facilities in the country. AERB now has obtained the necessary administrative and governmental sanctions for expanding the staff strength.	
79.	Switzerland	Article 8	“The Atomic Energy Commission (AEC) is a high level body dealing with policy matters concerning nuclear energy in the country. AERB enjoys full functional	For the regulatory activities, the financial budgets are prepared by AERB, which provides for establishment of infrastructure as well as sustenance of regulatory activities. The Budget of AERB forms part of the budget of the Central Government which is placed in	

			independence from DAE or any other agency in its functioning and its reporting to AEC is limited to presenting its Annual Report and Budget Proposals only once in a year." Question: Regarding the independence of AERB: Does AEC have to approve the budget proposal? What is the status of the Nuclear Safety Regulatory Authority (NSRA) Bill 2011? How does India plan to deal with the IRRS recommendation to secure the independence of regulatory body in the law?	the Parliament. The budget proposal is routed through AEC. Earlier the Government introduced the Nuclear Safety Regulatory Authority (NSRA) Bill, 2011 in the Parliament, with the aim of enhancing the existing 'de facto' independent status of AERB to 'de jure' independence. The NSRA Bill, 2011 could not be passed by the Indian Parliament before the term of the Lower House expired in 2014. Necessary administrative approvals are currently being obtained by the Government of India for re-introduction of the NSRA Bill in the Parliament.	
80.	United States of America	Article 8	One of the recommendations from the IRRS report was to further strengthen the existing legal and regulatory aspects regarding independence of AERB (securing the independence of the regulatory body in the law). What actions have been taken to address this recommendation?	Kindly see the answers to question no 71 and question no 78 posed by Switzerland under Article 7 and 8 respectively.	
81.	Canada	Article 8.1	Please describe what kind of licence (e.g. for siting, construction, commissioning, operation, decommissioning) exists, who must obtain the licence (e.g. NPP, manufacturer, designer,	Kindly note that section 8.1.1 deals with the "mandate and duties of AERB" and section 7.2.2.1 deals with the 'requirements and legal provisions of licensing under the Atomic Energy Act'. As mentioned in the last paragraph of section 7.2.2.2, the detailed consenting / licensing process is described under	

			operator, etc.) and who issues licences (e.g. AERB, other government bodies). As it is written in sub-article 8.1.1 it is not clear (III. Grant consents…).	Article 14 of the National Report. The specific details sought by Canada are brought out below. The regulatory system in India provides for issue of regulatory consents for NPPs for the stages (a) Siting, (b) Construction (c) Commissioning, (d) Operation and (e) Decommissioning. These consents are issued by AERB to the Utility.	
82.	Germany	Article 8.1	Could India please give a statement about the adequacy of your financial re-sources and how the total amount has developed during the last three years?	Yes, AERB has sufficient financial resources available for carrying out the planned activities. The total amount has been budgeted considering the expansion of India's nuclear program and expected increase in activities of AERB. AERB's activities are fully financed by the Government and the allocation for AERB has been increasing. Please refer to the earlier national reports of India to CNS.	
83.	Netherlands	Article 8.1	It seems that no research is done in the area of HOF. Why not?	The HOF related aspects get analysed implicitly as part of regular safety reviews and review of events. Please also see answer to Q. No. 7 posed by Canada under Article - General. The dedicated group in AERB for HOF is looking at HOF aspects more closely. The utility has an extensive arrangement of analysis of feedback from previous designs construction,commissioningand operation and incorporate the research done as part of OPEX.	
84.	Netherlands	Article 8.1	Many regulatory bodies in the world, face the challenge to transfer knowledge of retiring or	Kindly see answer to question no. 2 posed by Canada under Article – General.	

			senior staff to younger and/or new staff. Is this also the case in your country? Do you have a dedicated program for knowledge transfer and do you provide trainings to senior staff to improve their skills in knowledge transfer?		
85.	Netherlands	Article 8.1	The staff mainly consists of technical and scientific experts. On the other hand the majority of the safety incidents worldwide has its root cause in Human and Organizational Factors (HOF). To date many RBs have taken action and have recruited staff with for instance expertise in behaviour sciences or psychology. It seems this is not the case in India. Please explain.	Kindly see answer to question no 7 posed by Canada under Article – General.	
86.	Peru	Article 8.1	In the report, it is prescribed that Government had introduced the 'Nuclear Safety Regulatory Authority (NSRA) Bill 2011' in the Parliament with the objective of separation of primary legislation concerning regulation of nuclear and radiation facilities from other aspects. Which are those other aspects? Are these safety-related subjects?	As described in page 29 of the National Report, the legislative framework for all activities concerning atomic energy are governed by the Atomic Energy Act, 1962 and the Rules framed under it, provides for the development, control and use of atomic energy. The Atomic Energy Regulatory Board (AERB), the regulatory Body for nuclear and radiation safety is established by the Presidential Notification, using the provisions in selected sections of the Atomic Energy Act, 1962. The Nuclear Safety Regulatory Authority Bill, 2011 was introduced in the Parliament with the objective of	

				separation of primary legislation concerning regulation of nuclear and radiation facilities from other aspects in Atomic Energy Act, 1962.	
87.	Russian Federation	Article 8.1	It is stated in para 8.1 of the Report that Regulator utilizes the expertise available with three technical support organisations (Safety Research Institute, Bhabha Atomic Research Centre, Gandhi Centre for Atomic Research). Is there any split of areas of expertise among these organisations?	Safety Research Institute (SRI) is a part of regulatory body and carries out in-house research on areas of regulatory interest or limited scope independent safety analysis required for regulatory activities. Primarily, on generic safety research areas, LWR and PHWR related issues, the technical support is derived from BARC. In certain cases independent view of IGCAR is sought in the areas related to structural analysis, materials, corrosion, I&C aspects, NDT, etc.	
88.	Russian Federation	Article 8.1	Could you please present conclusions about the adequacy (or inadequacy) of human and financial resources of the regulatory authority.	Human and financial resources of the regulatory authority are adequate. Mapping of human resources and financial requirements is done for regulatory effectiveness and oversight vis. a vis. India's expanding nuclear power programme as well as the rise in the use of radioactive sources in Industrial, Medical and Research applications.	
89.	Germany	Article 8.2	Regarding the effective separation of the regulatory body you state that this has been ascertained by the IAEA-IRRS Mission in its report. However the IRRS report states in Recommendation 1: "The Government should embed in law, the AERB as an independent regulatory body separated from other entities having	It may be noted that AERB is established as a separate body with the necessary functional separation for effective independence. The observation of the IRRS Mission regarding this issue is brought out below. Quote: " The IRRS team noted the professionalism and integrity of the AEC, NPCIL and AERB senior staff	

			responsibilities or interests that could unduly influence its decision making.” Could India please elaborate how an effective separation of the regulatory body is achieved?	towards ensuring the regulatory decision making processes/arrangements were completed independently and did not notice instances, in which de-facto AERB independence was compromised. It was noted that the AERB has been established using the legal provisions of the AEA. With the statutory and legal provisions of the Atomic Energy Act and various rules framed thereunder and the powers conferred by its constitution, the AERB has the necessary legal authority for its regulatory activities. The mandate of the AERB doesn’t include any functions other than regulation of nuclear and radiation safety. These provide functional independence for the AERB as a regulator.” Unquote: The IRRS Mission, however, observed that that the regulatory body should be constituted through a legislative process thus demonstrating clear legal (de-jure) independence from the industry. However the IRRS Mission noted that while the AERB has necessary functional independence, there is potential for compromise, for which it recommended for embedding the ‘de-jure’ separation of the regulatory body in law. Further, please refer to answer for the question no. 71 posed by Switzerland on Article 7.	

90.	United Kingdom	Article 8.2	The report outlines how the work of the Regulatory Body is currently maintained separate from promotion activity as far as possible, despite both the industry and the Regulatory Body reporting within the same Atomic Energy Commission. Please explain how the process to introduce the proposed 'Nuclear Safety Regulatory Authority Bill' is being managed and what are the associated timescales (noting that the introduction of primary legislation to separate the regulation of nuclear and radiation facilities from other aspects would provide improved clear and transparent separation).	The NSRA Bill, 2011 could not be passed by the Indian Parliament before the term of the Lower House expired in 2014. Necessary administrative approvals are being obtained by the Government of India for re-introduction of the NSRA Bill in the Parliament. Kindly also see answer to the question no 88 posed by Germany under Article 8.2.	
91.	Germany	Article 9	How does the regulatory body determine the adequacy of the infrastructure of its on-site emergency?	Requirements with respect to on-site emergency are prescribed in AERB safety codes on design and operation of NPPs and the guides on Preparedness of the Operating Organisation for Handling Emergencies at Nuclear Power Plants (AERB/SG/O-6) and Criteria For Planning, Preparedness And Response For Nuclear Or Radiological Emergency (AERB/NRF/SG/EP-5 (Rev. 1)). Utility submits the information on provisions of required infrastructure for on-site emergency along	

				with their basis such as outcome of severe accident and study on effects of extreme external events. Establishment of adequate infrastructure for on-site emergency is a part of design safety review by AERB. Availability of the infrastructure is a prerequisite for regulatory clearance before initial fuel loading. During safety review, AERB confirms the adequacy of the infrastructure with respect to AERB safety codes and guides and in few cases independent confirmatory analyses are carried out. Further the outcome of emergency exercises as well as observations made during regulatory inspections are used to ensure that the requisite infrastructure is maintained.	
92.	Switzerland	Article 9	Both Indian licensees (NPCIL, BHAVINI) are fully owned by the Government of India. How does the Government of India guarantee that the regulator is effectively independent of the licensee (i.e. avoidance of regulatory capture) as Principle 2 of the IAEA Standards Series No. SF-1 is requiring?	Both Indian licensees (NPCIL, BHAVINI), reports to DAE, whereas AERB reports to AEC. So, functional independence is maintained. Please refer to the answer to question no. 88 posed by Germany Article 8.2.	
93.	Australia	Article 10	3rd bullet points states that each station conducts an annual safety culture survey but is this the same survey each year or is the survey process varied to prevent over-familiarisation by staff	Yes, it is the same survey each year. Safety culture assessment was introduced in the year 2015. Whenever the safety culture assessment system is revised, the aspects of safety culture survey to avoid over familiarisation will be taken care of.	
94.	Canada	Article 10	The following statement is found in the report: “Symptom based	a. The drawn inference is not correct. EOPs and SAMGs are different set of documents. As per	

			EOPs have been prepared and are under implementation." As NPPs should have had, since the beginning of their operation, the proper Emergency Operating Procedures (EOPs) to deal with abnormal incidents, can the Contracting Party clarify: a whether these EOPs are the equivalent of what is known as Severe Accident Management Guidelines (SAMG) in other jurisdictions b how the EOPs can be in both states: being prepared and under implementation c the completion date(s) of implementation.	established terminology for procedures available for operating personnel, EOPs are used to handle design basis accidents; whereas SAMGs/AMGs are for handling beyond design basis accidents. b. With reference to the quoted text, "Symptom based EOPs have been prepared and are under implementation", interpretation that (symptom based) EOPs are "being prepared" is not correct. Indian NPPs have well established practice of event based EOPs, which are used under transient and accident conditions (within the design bases). Scenario independent/symptom based EOPs have been prepared to complement event based EOPs. These symptom based procedures are advantageous in particular to handle multiple failure events. c. Symptom based EOPs are envisaged through a computer based system, which has been implemented in eight PHWR units, and is being improved based on feedback and modified system will be progressively implemented in PHWR units.	
95.	France	Article 10	India states that "AERB is developing safety performance indicators for measuring performance of the licensees, which are used as inputs for integrated assessment of the licensee's performance". Could India provide more details about these safety performance indicators?. Could India provide	Many utilities around the world utilize Safety Performance Indicators (SPIs) established by IAEA, in addition to other set of indicators specified by WANO. These indicators are intended primarily for use as a management tool by nuclear operating organizations to monitor their own performance and compare their performance globally. Such indicators while good for global comparison are not intended to identify the regulatory practices and required regulatory strategies to deal with specific problem to a	

			also a description of the integrated assessment process and how these indicators are expected to be integrated into this integrated assessment?	NPPs. In order to asses such issues, in-house development work towards realization of PIs at AERB has been initiated. A feasibility report was prepared and a pilot case was analyzed using the data obtained during the previous years. The framework was established for identification of defining safety performance indicators, data collection, assessment of the performance indicators, and preparation of regular reports. Among other things, PI methodology accounts for following factors namely: Significant events and their reporting, adherence with Technical Specifications, status of Radiation Protection, assessment of Nuclear Safety, findings of Regulatory Inspections and their resolution and Safety Review findings. Based on the assessment, PIs are evaluated for each NPP. The output thus generated is utilized in prioritizing the regulatory attention to the generic safety/ safety cultural issues as well as on specific issues of an individual NPP.	
96.	Pakistan	Article 10	It is stated that AERB is developing safety performance indicators for measuring performance of the licensees which are used as inputs for integrated assessment of the licensee's performance. India may share the list of these safety performance indicators and the basis of their selection?	Kindly refer to the answer for question no 94 posed by France under Article 10.	
97.	Peru	Article 10	In the report is prescribed that Safety Culture attributes have been	The attributes selected for assessment of safety culture of the Regulatory body and the safety culture	

			adopted from the international guidelines and modified to suit the AERB requirements. Are those attributes or characteristics consistent with those recommended by IAEA (GS-G-3.1)?	assessment of operating NPPs are based on various international practices and guidance including OECD-NEA, IAEA GS-G 3.1 and other country practices.	
98.	Russian Federation	Article 10	How does India implement Vienna Declaration on Nuclear Safety principle that national requirements and regulations on safety culture should take into account relevant IAEA Safety Standards?	All the national safety requirements / regulations for Indian NPPs take account of the relevant IAEA standards, including for safety culture. The AERB Code (AERB/NPP/SC/QA(Rev.1) spells out national requirements on safety culture. These are being reviewed in relation to GSR – Part 2 issued by IAEA recently.	
99.	Russian Federation	Article 10	Could you please give key results of periodic internal and external assessments of safety culture.	Internal safety culture assessment is performed at stations as per NPCIL Headquarter Instruction - 0559 “Assessment and Fostering of Safety Culture at Nuclear Power Stations”. External safety culture assessments are performed as a part of WANO peer review of stations. Some of the elements identified for improvement pertain to following principles of safety culture are: i) Leaders demonstrate commitment for safety. ii) A questioning attitude is cultivated. In general, safety culture assessment results of all stations are healthy.	

100.	Switzerland	Article 10	The report states that the Regulator has formulated Safety Codes specifying detailed safety requirements for the NPPs. These Codes require, among others, that the utility shall ensure that safety culture and that plant management shall inculcate safety culture in plant personnel. Could you please outline the AERP understanding of safety culture? What methods are best suitable to inculcate the AERP understanding of safety culture?	The AERB understanding of the safety culture is the same as that of the IAEA understanding as also reflected in the AERB documents related to safety culture. AERB encourages every utility to institute a good safety culture during all the stages including design, construction, as well as operation of an NPP. The regulatory requirement for establishing safety culture within utility is delineated in the AERB safety code for quality assurance in nuclear power plant- AERB/SC/QA (Rev.1) and related guides. The review and assessment of the safety culture is also a part of AERB's continual safety review through a multi-tier review mechanism. Continual safety review involves extensive interactions with plant, personnel and management which provide opportunity for the regulators to assess the broader perspective on the safety culture prevailing at the NPP. While taking a regulatory decision this perception is also used along with the technical results.	
101.	Switzerland	Article 10	The report states that all the nuclear power stations of NPCIL have established safety culture assessment. For this AERB has developed a safety culture assessment system to inspect and recognize early symptoms/signs of declining safety culture of the utilities. Could you please give	For the independent assessment of Safety culture, AERB has developed its own methodology based on various international guidance available. The early signs of declining safety culture have already been identified by OECD NEA in its document "Improving nuclear regulation" and the complete list of such symptoms is available in the document. Following are some attributes against	

			examples of already recognized early symptoms/signs of declining safety culture when conducting inspections at the utilities?	which safety culture is assessed: 1. Frequent deferral of needed improvements 2. Long delays to meet regulatory commitments	
102.	Switzerland	Article 10	The report states that AERB has initiated a process for assessing the safety culture of itself and that based on these assessments, management actions are taken. Could you please outline the process that AERB has initiated for assessing its own safety culture as well as give examples for management actions that AERB has taken to enhance its safety culture.	AERB conducts a safety culture survey amongst its employees on an yearly basis. The survey results are analysed and mapped to the already established safety culture attributes. If the result of the analysis shows degrading trend in any of its attribute, management action is initiated regarding the same. As an example, the transparency was increased between the employees by establishing a clearer reporting structure and the job allocation of the employees.	
103.	Canada	Article 11	The report states that “Minimum staff requirements are met as a part of Limiting Conditions of Operation (Technical Specifications for Operation) and any non-compliance may attract the regulatory enforcement. ” What type of regulatory enforcement due to non-compliance of such nature does AERB administer?	The technical specifications for operation of NPP specifies requirement for minimum staff at the plant on shift basis. The NPPs have additional crews of trained and qualified personnel. With this arrangement, we haven’t faced situations involving non-compliance to minimum staff requirement at any of the NPPs. However, if such a scenario arises, utility is required to shut-down the reactor as limiting condition for operation is not fulfilled. In case of non-compliance, the regulatory body will issue necessary directives to ensure the same.	
104.	Canada	Article 11	Please clarify: a Who evaluates and qualifies	a. The Contractors are evaluated and qualified by the licensee.	

			contractors: licensee or AERB? b Do contractors need a licence for specific work?	b. The contractors are evaluated through a Vendor Evaluation Criterion established by the Licensee. However, no license is issued to the contractors for specific work.	
105.	Netherlands	Article 11	How does the regulatory body assess the sufficiency of human and financial resources at the nuclear installations?	The regulatory requirements as regards to human resources at nuclear installations are detailed in Sections 11.2.1 to 11.2.7 of the National Report. AERB has specified a detailed set of regulatory requirements concerning the human resources at the nuclear installations in the AERB Safety Code on NPP Operation (AERB/AERB/NPP/SC/O Rev.1-2008), AERB Safety Code on Quality Assurance in NPPs (AERB/NPP/SC/QA Rev.1– 2009) and in AERB Safety Guide Staffing, Recruitment, Training, Qualification And Certification of Operating Personnel of NPPs. AERB's assessment of aspects related to human resources are carried out at the time of initial licensing (based on which the LCOs are included in the Technical Specifications for Operation of the NPPs) as well as during the PSRs. These aspects are brought out in the National Report in sections 11.2.7 and 11.2.9. In India, NPPs are allowed to be established and operated only by the Government or by an authority or corporation established by it or a Government Company. Presently there is one utility in India operating NPPs (NPCIL). The financial resources of NPCIL come from budgetary support from Government of India, borrowings from capital market	

				and internal surpluses. NPPs are allowed operate only if the safety / regulatory requirements are fulfilled, irrespective of the cost involvement. The necessary financial resources for management of a radiological emergency will be made available by the Government. The regulatory body does not specifically assess the financial provisions for this purpose. There is a separate provision for a decommissioning reserve established by the Government.	
106.	Switzerland	Article 11	The report does not mention if shift and maintenance personnel of NPPs undergoes regular training concerning Human Performance Tools The number of four simulators to train the operators of 21 NPPs seems rather low. How is it assured that all licenced personnel gets adequate simulator training?	Effective utilisation of Human Performance Tools by shift and maintenance personnel of NPPs is checked regularly as a part of Job Observation Program and the performers are coached, if any gaps are observed. In addition periodic as well as need based, class room cum demonstration trainings are conducted on effective use of Human Performance Tools. NPCIL follows twin unit design concept, in which two units with same design constitute a station. There are 5 simulators in 11 stations. As per the procedure approved by the regulator, personnel of stations which have simulator, undergo initial simulator training for 6 weeks and periodic simulator training for one week every year on their respective plant simulator. The personnel of stations that do not have plant simulator, undergo initial simulator training for six weeks at the simulator of similar design at other station, and in place of periodic simulator training, the intent of plant	

				simulator training is met through refresher of procedures and group discussions at training centres and by a virtual enactment of the procedures by demonstrating action steps in a sequential manner in front of control panels and by a mock role play by team members.	
107.	Czech Republic	Article 11.1	The chapter does not provide a description of the Contracting Party's arrangements for ensuring that the necessary financial resources are available in the event of a radiological emergency. Are there any such arrangements?	The central and state governments provide funds for immediate relief and rehabilitation to address the needs of the affected population in case of a radiological emergency.	
108.	Germany	Article 11.1	Regarding the financial resources of the licence holder could India please provide a statement to the adequacy of financial provisions, the regulatory body's processes to assess the financial provisions and a description of Indias arrangements for ensuring that the necessary financial resources are available in the event of a radiological emergency.	Please see India's responses to Question No. 104 posed by Netherlands and Question No. 106 posed by Czech Republic under Article 11.	
109.	Peru	Article 11.1	In the report is prescribed that the management of all NPPs prepare a list of safety culture indicator as applicable to their site. Does it mean that indicators may vary from one site to another?	These indicators are prepared by utility and are same at all stations. These indicators are related to the self-assessment of the safety culture by the utilities and are not proposed or recommended by the AERB. However, the	

			Are these indicators proposed or recommended by regulatory body?	indicators and the results are reviewed by the AERB during the regulatory inspection of that NPP. Further, AERB has developed an independent assessment methodology for assessing the safety culture of the utilities which is also described in the national report.	
110.	Australia	Article 11.2	This section indicates that in an emergency at one unit on a two (or more) unit site, staff from the unaffected unit can supplement the staff at the affected unit. However, what happens if there is an extreme external event that impacts both units simultaneously? It is noted that this issue is discussed further in section 19.4 but the issue of simultaneous events on multiple units is not really addressed.	Indian NPPs follow twin unit station concept i.e. two NPP units constitute a station. Stations have their own independent staff for operation and emergency handling. The said statement indicates that such provision (augmentation of the staff at the affected unit from the unaffected one) may be utilized if the situation demands during an unforeseen accident in single unit at a multi-unit site. However, in case any extreme eventuality requires augmentation of the staff available at the site, personnel from other NPPs/ HQ can be deployed for the required duration.	
111.	France	Article 11.2	India states that “Contractor’s personnel are not allowed to carry out any job without supervision. They are not deployed for carrying out any operations in the control room and vital areas.” Could India describe the approach used by AERB for assessing and verifying that the license holders have appropriate provisions (e.g. staffing, competencies, procedures,	In Indian NPPs, contractors are not employed for routine operation in critical areas of the main plant. The contractors are restricted to carry out operational activities in the auxiliary facilities like switch yard, DM water plant, chiller plant, etc. During the biennial maintenance shutdowns, contractor’s manpower is used to supplement the plant personnel. In this period, the contractor’s personnel work alongside and under the supervision of the regular plant personnel and no independent responsibilities are assigned to them. Such personnel are provided specified training,	

			etc.) for carrying out an efficient supervision of contractors in the field, in particular when tasks performed by contractors are important for safety?	including radiation protection. AERB requires the licensee organisation to establish, implement, assess and continually improve a detailed QA programme, to demonstrate that the programme is consistent with the regulatory requirements, for the life cycle of NPP. The programme outlines the special requirements necessary to effectively manage the processes carried out in multiple organisational arrangements such as contractors, sub-contractors and functional units within an organisation. This QA programme is reviewed and approved by AERB as part of the application for license. The licensee has the responsibility to make proper arrangements with vendor(s) and/or contractor(s) availability of all the required information and also keep the regulatory body constantly informed of all relevant additional information or changes in the information submitted earlier. The licensee is also required to ensure that the consultants and contractors that carry out assignments and activities also follow the safety and quality assurance norms of the licensee. The Contractors are evaluated through a Vendor Evaluation Criterion established by the licensee. In the field before undertaking actual work, contractor personnel are given appropriate training, briefing and are provided with approved work procedure. The work is carried out by the contractor, under the supervision of licensee’s personnel. QA checks and critical checks are done by the licensee.	

				AERB verifies the aspects related to adherence to the QA programme including related documentation, as part of the inspections, safety assessments and verification of the licensees.	
112.	Germany	Article 11.2	Based on INFCIRC/572/Rev.5 Article 11 (2) bullet 11 could India please describe which methods India uses to analyse the competence, availability and sufficiency of the additional staff that is required for severe accident management, including contracted personnel or personnel from other nuclear installations?	Subsequent to preparation of accident management guidelines at NPPs, all licensed and qualified personnel undergo periodic training on accident management. Periodic drills are also carried out in which usage of accident management measures are rehearsed, which involve assessment of competence and sufficiency of additional staff required. Requirement of contractor personnel is not envisaged in accident management guidelines. Immediate actions are envisaged to be taken by the staff of the affected NPP. As accident management philosophy is same across the fleet of reactors, personnel from other NPPs can also provide help in case such a need arises. In this context it is worthwhile to mention that all NPPs in India are operated by the same utility and therefore getting help from other NPPs is easily manageable.	
113.	Canada	Article 12	The report states that nuclear installations are operated within the limits specified in technical specifications. Can the Contracting Party clarify if beyond-design-basis operating guides (e.g., SAMGs) have been established?	The section quoted in the question (#12.2.3) deals with normal operation. Yes, accident management guidelines (SAMGs) have been established and information of these can be found in Article 19.4 (Page 166-167) of the national report.	
114.	Canada	Article 12	According to the report, design is “aimed at limiting the effects of	Maintenance activities are guided by maintenance procedure & checklists. During performance of	

			human errors during normal operating conditions, transients and during maintenance.” Maintenance is often the area where events and incidents with human factors implications occur. During maintenance activities, what human factors processes are used to ensure that human errors are kept low?	maintenance activities, human error prevention tools like pre-job briefing, adherence to procedures & checklists, job-site review, hold points, independent verification, Foreign Material Exclusion , JIT briefing are used. Also regular training on mock-up facilities for critical activities is imparted to maintenance personal to preclude human errors. Additionally, in all the stations Job Observation programme has been implemented. Job observation team observe the conduct of maintenance activity with respect to pre-job briefing, adherence to maintenance procedure, job-site review, flagging, post job debriefs, etc. For gaps observed, if any, with respect to desired behaviour, the concerned job performers are coached accordingly.	
115.	China	Article 12	According to Article 12 in the guidelines INFCIRC 572, “Methods and programmes of the licence holder for analysing, preventing, detecting and correcting human errors in the operation and maintenance of nuclear installations” should be included, Question: How to analysis, prevent, detect and correct human error in the operation and maintenance of NPPs in India?	The details addressing human performance issue during operation and maintenance of NPPs are elaborated in 12.2.3, 12.2.5& 12.2.6/P87 of the report. Further, a Human Performance Enhancement programme has been implemented at all stations based on the guidelines given in Head Quarter Instruction (HQI)-0550 (R-0). As per the requirements given in the HQI, the sectional and station level human performance coordinators identify the human performance related issues through various station programmes during operation and maintenance of NPPs and discuss in the meetings of Human Performance Review & Enhancement Committee (HUREC). Also the root cause analysis of the events is carried out based on the guidelines given in Head	

				Quarter Instruction (HQI)-0449 (R-0). This HQI provides the methodology to identify if the event has taken place due to human error and also identify the failed barriers. Once the failed barriers are identified, appropriate actions are taken to avoid recurrence of human errors in operations and maintenance.	
116.	Czech Republic	Article 12	Which HRA method is used to support the PSA model?	Technique for human error rate prediction (THERP) is used to model latent human actions. Dynamic human actions are modelled by using Human Cognitive Reliability (HCR) model for diagnosis error and accident sequence evaluation program (ASEP) for execution error	
117.	France	Article 12	Could India precise how many HOF specialists are working in AERB for taking in charge all issues related to human factors? What are the requirements (such as background, competencies, experience and others) expected from a HOF specialist? How are their roles and responsibilities defined? Does AERB rely on support from external HOF specialists (contractors, academics, etc.)?	As explained in answer to question no.7 posed by Canada, under article – General, AERB is in the process of developing full time dedicated competencies in soft skills, including HOF. Even though this work is in hand as suggested by the IRRS Mission, AERB has sufficient number of personnel who have the necessary skill and experience in dealing with HOF aspects including safety culture, HMI, to carry out its review and assessment of HOF aspects vis-à-vis the established requirements. AERB has the necessary provisions / powers to engage the specialists in these areas, if found necessary, by engaging such specialists as consultants or as members in the committees. At present such specialist have been engaged to provide HOF related training to senior management personnel. It is expected from the HOF specialist within AERB that he/she should have adequate knowledge about	

				NPP system design, safety and regulatory requirements, experience in safety review and assessment, understanding of NPP design & operational aspects, organisational interfaces, safety culture aspects, etc., so that he/she should be able to analyse consequences of the probable errors/ unsafe conditions / acts, while performance of the tasks. Roles and responsibilities of such specialists would include review and assessments, effective interface for identifying HOF issues having safety implications and facilitate their resolution through suitable changes in different regulatory processes.	
118.	France	Article 12	Could India precise if AERB has any requirement or expectation about the presence and activities of HOF specialists in license holder staff?	The present regulatory requirement of AERB does not specifically require recruitment of HOF in the license holder staff. Section 12.1 of the national report discusses the AERB requirements with respect to the human and organisational factors and 12.2 of the national report brings out the considerations for human factors in different activities during the life time of NPP. Excerpts from Section 12.1 of the report dealing with regulatory requirements on human factors is given below. The AERB Safety Code on Quality Assurance in NPPs (AERB/SC/QA, Rev1, 2009) covers the senior managerial commitment to foster involvement of all in organisational QA aspects and safety culture.	

				AERB Safety Codes on Design of PHWR based NPPs, AERB/SC/D (Rev.1, 2009) and Design of LWR based NPPs AERB/NPP-LWR/SC/D (Rev-0, 2015) specify requirements for design of NPPs for optimised operator performance. The requirements cover need for designing working areas and environment according to ergonomic principles, systematic consideration of human factors and the man-machine interface, etc. AERB Safety Code on Nuclear Power Plant Operation (AERB/SC/O, Rev.1,2008) gives requirements related to reducing the human errors. The AERB Safety Guides on Safety Related Instrumentation and Control for Pressurised Heavy Water Reactor Based Nuclear Power Plants (AERB/SG/D-20) and Radiation Protection in Design (AERB/SG/D-12) provide detailed guidance on design for optimum human performance. AERB document on ‘Human reliability analysis (methods, data and event studies) for NPPs’ (AERB/NPP/TD/O-2) provides various methods and illustrative examples for estimation of human error probabilities. Accordingly Quality Management System (QMS) principles are practiced in Nuclear Power Corporation of India Limited (NPCIL). A formalized system has been established and documented in “Corporate Management System Document Rev.2, March 2015”, based on the AERB Safety Code on Quality Assurance and international documents (e.g. IAEA GS-R-3) on the subject. This document encompasses policy, organizational arrangements, roles/ responsibilities and related measures to be	

				implemented during all stages of NPPs as applicable to the activities and functions of NPCIL engineering activities and NPP Sites.	
119.	France	Article 12	India states that "Availability of a training simulator is a mandatory regulatory requirement for licensing of NPP", and also "Special training courses are also arranged for all the concerned personnel on the design changes that are carried out.". Could India precise if the training simulator is a full scope simulator and if it is used not only for training, but also for running tests for validation of the design, in particular the integrated system validation (ISV), in case of new build as well as modification (for instance, refurbishment of the main control room) ? Could India describe how AERB experts are associated in the elaboration of the test scenarios, do they have possibility to collect data (observations of tests, interviews of personnel involved, etc.), and on the basis of which criteria they analyze the results of the validation tests?	(i) NPP Simulators in India are real time replica full scope training simulators (FSTS) used for training of candidate control room engineers and also for re-training of qualified operators. The FSTSs are kept updated to reflect changes in Main Control Room (MCR) due to modifications in plant systems. (ii) These simulators apart from operator training are also used for- a. Validating new plant designs and selected systems. b. Recreating plant occurrences from time to time, to support analysis of the cause of such an occurrence. (iii) AERB observes simulators functioning as part of regulatory inspections and they have access to all information related to FSTSs.	
120.	Switzerland	Article 12	The report states that AERB Safety Codes on Design of PHWR based	The term 'Systematic consideration' refers to a system in which considerations for HOF aspects and	

			NPPs and Design of LWR based NPPs, inter-alia establishes the requirements for design for optimized operator performance. These requirements include the designing working areas and environment according to ergonomic principles, a systematic consideration of human factors and the man-machine interface. Would you please outline your concept of a systematic consideration of human factors (methods, criteria, indicators, etc.) for the design of NPPs?	influencing factors are given during all stages of the NPPs and involved processes. This includes SSCs of NPP ergonomically designed to address human-machine interface issues, operators are competent to perform assigned tasks (academic qualification, training and livening/ certification) using approved and validated procedures. Key positions of the operating staff undergo simulator training and evaluation to check their response prior to issuance of licence. All these steps which are subjected to multi-tier review and assessment. Concept of systematic approach includes provisions for consideration of the human interface with technology, organization i.e. clarity in defining roles/ responsibilities/ authorities and the work procedures/ instructions for performance of specific jobs and the environment. Event reports and near-misses due to unsafe act can be considered as indicators. The systematic approach to ensure optimised operator performance initiates right at the design activities taken up by NPCIL through obtaining and implementing feed-back from experienced operations and maintenance staff. Similarly for the construction activities, feed-back from earlier construction activities as well as reputed contractors is considered while finalising the design. Similar aspects apply to commissioning, training programmes and operation.	
121.	Switzerland	Article 12	The report states that organizational factors and	Please also refer answer to Q. No. 7 posed by Canada under Article – General.	

			managerial aspects have a major impact on the behavior of individuals. AERB Safety Code on Quality Assurance in NPPs covers the managerial commitment to improve human factors to enhance safety in NPPs. This Code requires that management shall determine the competence requirements for individuals at all safety levels and shall provide training or take other actions to achieve the required level of competence. Are there, besides training to achieve the required competence, other managerial aspects that have a major impact on the behavior of individuals that AERP is addressing in its oversight activities? How the term “human factors” is understood in the context of the AERB Safety Code on Quality Assurance in NPPs.	The human factors refer to factors which have significant influence, in a positive or adverse manner, on human performance. These factors are having interactions with and may get affected by the organizational, technological and environmental factors including human attributes (knowledge, skill, fitness, attitude and motivation etc.). Other external aspects (e.g. social, political etc.) beyond control of the licensee are not covered in the regulatory oversight. In addition to training, qualification, licensing and authorization, regulatory oversight cover the other managerial aspects such as requirements related to well defined organization structure, clarity in role and responsibilities of key positions who are part of decision making process, authorities to be commensurate with responsibilities, and freedom and ability to take decision on the matters/ actions needed to ensure safety. AERB requires that management has to ensure that all the activities are properly planned defining logical sequence based on their interfaces/ interactions, necessary resources are provided to be executed by competent and authorized personnel, assessment of the results for meeting the intended objectives and identification of further improvements. AERB reviews cover among others, the organizational structure of NPPs, roles and responsibilities and verification of effectiveness of the performance during all stages of NPPs. Safety and quality issues are given overriding priority over all other requirements.	

122.	Australia	Article 13	Is it intended that all directorates at NPCIL will eventually be subject to ISO 9001: 2008 certification? In addition, are there plans to upgrade the current directorate certifications from ISO 9001: 2008 to ISO 9001: 2015?	Most of the directorates in NPCIL are already certified for ISO 9001:2008. Yes, it is planned to upgrade the current directorates' certification to ISO 9001:2015 before the end of year 2018.	
123.	Australia	Article 13	This section only appears to address NPCIL QA systems. For completeness, it should also cover the Regulatory Body's QA system. It is noted that the Regulatory Body's QA is summarised briefly in section 8.1.2.8 but not to the depth required under Article 13.	As per the Guidelines regarding National Reports under CNS (INFCIRC/572/Rev.5), the section under Article 13 is to address the management system of the Licensee.	
124.	Australia	Article 13	Are the internal audits carried by certified auditors within NPCIL?	Yes. Qualified internal auditors carry out audits within NPCIL plants and at head quarter for ISO 14001, IS-18001, ISO-17025, ISO-9001.	
125.	Canada	Article 13	It is mentioned that procurement of structures, systems and components is made from duly qualified and approved suppliers, and that they meet the applicable regulatory, statutory and other stated requirements specified in the Procurement Document(s). Who is responsible for ensuring that the quality programs for goods and services supplied by subcontractors	The utility is responsible for ensuring the quality programme for goods and services supplied by contractors/sub-contractors to meet the nuclear power plant requirement.	

			meet the nuclear power plant requirements?		
126.	Switzerland	Article 13	Requirements, sequence and interaction of processes and activities, criteria and methods needed for implementation and control, process inputs and outputs are specified and their effectiveness is ensured How is the effectivness of the Management System being controlled?	As per the Guidelines regarding National Reports under CNS (INFCIRC/572/Rev.5), the section under Article 13 is to address the management system of the Licensee and accordingly the same has been covered in detail.	
127.	Switzerland	Article 13	Graded Approach: Management System Programm has provision for such graded approach for different processes, items and services. Does this mean that the processes of the Management System are categorisied according to their safety relevance?	Processes of the management system are generally not categorized. However, graded approach is applied in each activity of different processes and activities depending up on their significance with considerations as specified in the regulatory documents and the established management system. The graded approach aims to have planned and recognised difference in the application of specific QA requirements. Nuclear Safety is the fundamental consideration in the identification of items, activities. Whilst QA principles remain the same, the extent to which QA requirements are to be applied are consistent with the importance to nuclear safety of the items, activities of the processes.	
128.	Switzerland	Article 13	Interface Arrangements: Functional interfacing and cross-functional integration of core processes i.e. Siting, Design,	Yes, NPCIL has one overall Management System, governed by a Policy document titled as "Corporate Management System Document (CMSD)". Based on this document, each Directorate at HQ prepares	

			Procurement, Manufacture, Construction, Commissioning, Operations and De-commissioning and also the supporting processes are implemented in a coherent manner to meet the necessary agreed arrangements and responsibilities. What about the scope of the Management System of the licensee holder? Does the NPCIL have one overall Management System with appropriate interface arrangements for the different activities (Quality assurance programms) or are there several MS in place according to different sites and NPP life time stages?	interface document in line with the CMSD. Similarly, all the projects and stations of NPCIL has Quality Management System documents titled "Quality Assurance (QA) Manual" interfacing different activities in line CMSD.	
129.	Switzerland	Article 13	Measures for continuous improvement are initiated in the management system accordingly. How is the effectivness control of the CI measures in the frame of the PDCA cycle (Management review) practically implemented?	As indicated in Article 13.4 of the report, mechanism for continual improvement in PDCA cycle is part of the management system of the organization and QA program for its processes. All activities are planned, performed and assessed as per approved procedures to achieve the set objectives. These aspects are monitored and assessed throughout their execution. Deviations, if any, are analyzed for required corrective and preventive actions. Verification of compliance with the regulatory and safety requirements is done by AERB through safety review performed for licensing/ authorization and	

				regulatory inspections. The Senior Management of the organization identifies, prevents and corrects the problems that hinder achievement of the specified objectives. Self-assessment at all levels and independent assessment is considered to be effective tools to achieve these objectives. All the Managers and Task Performers periodically evaluate their work to compare current performance to expectations in respect of worldwide industry standards of excellence (bench marking), meeting stakeholder requirements and expectations, regulatory and statutory requirements, and to identify areas based on experience, feedback and lessons learned from incidents or any other inputs received needing improvement in all stages of PDCA cycle.	
130.	Canada	Article 14	The following statement is found in the report: "…such as installation of hydrogen management provisions, provisions for containment filtered venting system and creation of on-site emergency support centre are also in progress." As the Fukushima accident occurred more than five years ago, can the Contracting Party clarify: a whether the statement above means that the NPPs are only now being retrofitted with hydrogen igniters and passive autocatalytic	a) As explained in the answer to question no 56 from Switzerland under Article 6, the provisions for handling severe accident were under development for Indian reactors well before the Fukushima accident (ref pages 16 & 22 of the Indian National Report for the 5th Review Meeting of CNS, submitted in August 2010). Certain inherent design features available in the Indian standardised PHWRs (large water inventory in calandria and calandria vault, large containment volume, etc.) provide relatively large time for the accident mitigating actions. The AERB Safety Code on Design of PHWR based NPPs AERB/NPP-PHWR/SC/D, published in 2009, incorporated additional requirements related to accident management; and the development work for strengthening accident management provisions in the	» Note on Features of Indian PHWRs

			recombiners b whether a containment filtered venting system will be installed in every site c whether the Emergency Support Centre will be servicing all the NPPs in India d the completion dates for the above	existing reactors as per a technical basis and provision for passive autocatalytic recombiner devises for strengthening the defences against hydrogen flammability, etc. were in hand. The Fukushima accident further prompted for expeditious development and enhancement of measures related to SAM. b) The need of CFVS (Containment Filtered Venting System) has been finalized and will be provided in the plants, as required, based on the accident analysis. It may be noted that as explained in the answer to question no 197 posed by Canada under Article 9, from the accident analysis carried out for PHWRs of lower capacity and large containment volume, it is seen that the containment pressure remains within its design pressure for 7 days into the accident. This time is considered adequate to make alternate provisions for containment cooling. (Please refer attachment titled “Note on Features of Indian PHWRs”). c). One OESC (On-site Emergency Support Center) will be set up at each NPP site, which will cater to the emergency requirements under severe accident condition in one or multiple units of that site. OESC of different sites are independent from each other. d). All these measures are part of the long-term actions as categorised under the action plan for the post Fukushima Safety enhancements. As explained in the answer to question no 3 posed by Canada under Article – General, these measures have been initiated	

				for implementation, to be completed in a phased manner over next two years.	
131.	Canada	Article 14	Section iv) describes the basis for licensee aging management programs for SCCs important to safety. In developing Safety Guide AERB/NPP/SG/O-14, to what extent were staff guided by the following IAEA documents: • Safety Guide NS-G-2.12, "Ageing Management for Nuclear Power Plants; and, • Safety Report Series No. 57, "Safe Long Term Operation of Nuclear Power Plants".	AERB/NPP/SG/O-14 is a safety guide issued in 2005. In review process of ageing management programmes at NPPs, this guide along with the current IAEA documents, including NS-G-2.12, are used.	
132.	Switzerland	Article 14	What is the legal and administrative measure basis to guarantee independence of the regulator for the assessment process? How is the supervision during erection and plant operation organized between this organization and AERB?	As explained in detail in the National Report under Article 7, AERB has been vested with the necessary legal authority / powers for specifying safety and regulatory requirements for regulation of nuclear and radiation facilities / activities, issuance of regulatory consents, conduct of safety reviews / verification and to take enforcement actions, under section 17 and 23 of the Atomic Energy Act, 1962 and the Rules made thereunder. During all life cycle of NPP starting from siting, design, construction, commissioning and operation regulatory. The interfacing between AERB and the licensee are that of the Regulatory Authority and the Licensee.	

				Further, please refer answer to question no. 88 posed by Germany on Article 8.2.	
133.	Switzerland	Article 14	Is there a centralized storage facility for accident management equipment or does a specialized crisis management team / nuclear rapid response force exist?	Each NPP has an independent centralised storage facility for accident management equipment. An emergency organisation team exists at each NPP for handling different kinds of emergencies (plant /site / off-site). The roles and responsibilities of each individual / group are well defined in the respective emergency preparedness and response manual. For details of management for onsite and offsite emergency kindly refer article-16 of Indian National Report.	
134.	Switzerland	Article 14	Which role does PSA play the in safety assessments of nuclear installations? To which extent is PSA applied in the safety assessments of nuclear installations?	PSA has a complementary role to deterministic safety analysis in the safety assessment of the NPPs. The PSAs are required to be updated taking into account of design/procedural modifications and component failure data. The PSA results are presented as a part of periodic safety review (PSR), which is conducted every 10 years. The PSA results are considered suitably in regulatory decision-making for additional insights, along with the outcome of deterministic safety analysis and other safety assessments.	
135.	Switzerland	Article 14	Usually, TSO’s can work for both the regulator and operator. How is	In regulatory decision making, the responsibility for safety assessment and regulatory decision making are solely with AERB. The TSO’s support is used in	

			the independence of TSO's and their expertise ensured?	conduct of the safety reviews and inputs from the TSO forms one of the inputs for the safety assessment. AERB has established a formal MoU with BARC, for technical support, which incorporates the obligation of promptly notifying AERB, in case of any conflict of interest aspects are identified with respect to any individual expert providing technical support to AERB.	
136.	Switzerland	Article 14	The continued verification of safety includes following programmes: Surveillance, In-service Inspection, Maintenance, Establishment of programme related to life management, Performance Review, and Update Probabilistic Safety Assessment. How are deterministic aspects being addressed in the continued verification process, .e.g. hazard re-evaluation due to new findings, an update of the safety analysis, a use of new computer codes in order to achieve state-of-the-art conformity?	The continued programs for verification of safety, such as surveillance, in-service inspection, life management and performance reviews of systems, structures and components (SSCs) important to safety confirms that NPPs remains within the assumptions considered in the safety analysis. If the condition of any of the SSCs is found to be outside these assumptions, the safety analysis is carried out with the as found condition of these SSCs. As a part of Periodic Safety Review, the safety analysis (both deterministic and probabilistic) is reviewed in comparison to current requirements and practices, with respect to analytical methodologies, modelling, consideration of PIEs, assumptions used, conservatism / uncertainties, etc. among a number of aspects. This also includes any change in hazard evaluation due to new findings. The purpose of the review and assessment are to see the need for any revisions in the analyses and to ensure continued compliance with the requirements. Revisions in the analyses are mandated if found necessary.	

137.	Switzerland	Article 14	How are the PSA (probabilistic safety analysis) and DSA (deterministic safety analysis) connected / interacting in the decision-making process?	Kindly see the answer to Question no 133 posed by Switzerland under Article – 14.	
138.	United Kingdom	Article 14	The National Report section 14, specifically section 14.1.2 ‘assessment of safety through the licensing process’ states the Federal Authority for Nuclear Regulation (FANR) has established in its management system a process consistent with the Nuclear Law and the relevant IAEA safety requirements for assessing applications for licences relating to the construction and operation of a nuclear facility. Please provide further information on how UAE’s obsolescence management process of Instrumentation and Control (I&C) for equipment has been implemented throughout the design, manufacture & procurement, installation and commissioning lifecycle.	The question is not related to the Indian National Report.	
139.	United States of America	Article 14	The report for the renewal of a license is submitted to AERB three months prior to the expiration of	The established regulatory requirements don’t prescribe the conduct of regulatory inspection as an essential part of the license renewal process. The PSR	

			the operating license. AERB conducts a detailed review of the report and issues the license after being satisfied that the plan could be operated in a safe manner. Does AERB performs inspections as part of the license renewal process?	methodology has been described in 3rd para of page 15 and section 14.1.2.5(ii) of the Indian National Report; which comprehensively cover all aspects related to assurance of safety, including resolution of the findings of regulatory inspections conducted. However, AERB is free to carry out regulatory inspections in response to issues emanating from the outcome of a safety review (e.g. safety review as a part of continual safety supervision of NPPs) which may or may not be directed towards license renewal. Detailed description of the regulatory inspection practices in India is given in section 14.2.3 of the National Report (page 107 & 108).	
140.	Australia	Article 14.1	It is not clear in the description provided what is the difference between the 5-yearly periodic safety assessments and the 10-yearly PSRs? Some clarification of the differences would be beneficial to understanding.	As per the current practice followed in India, renewal of license of operating NPPs is granted for a maximum period of five years. These license renewals are based on a safety review. It may be noted that the scope of two consecutive safety reviews are different i.e. one is a comprehensive PSR the other one being of a limited scope of review. License renewal for operation of NPP in every 5 years is a regulatory requirement wherein utility is required to submit application in a prescribed format, covering details on safety factors such as operational safety performance, operational experience feedback, actual physical conditions and public concern. PSR is more comprehensive review during which, in addition to the above safety factors, improvement in safety standards and operating practices, cumulative	

				effects of plant aging, plant modifications, safety analysis, etc. are also considered. The key aspect of the PSR is that it involves assessment of the safety factors of the NPP in comparison with the current safety requirements and practices. Based on this assessment, strengths of the NPP and need for safety enhancements are identified.	
141.	Australia	Article 14.1	Item ii indicates that any modifications to safety or safety-related systems are subject to regulatory review and approval but is this regardless of the safety significance of the modification itself?	Yes, any design modification in the safety and safety related systems require regulatory review and approval. The effects of these modifications on safety functions are independently assessed in the regulatory body. The review process in practical terms is commensurate with safety significance of the modification	
142.	China	Article 14.1	Question: Is the Safety Review and licensing for standardized NPP design included in Indian nuclear regulatory body? If yes, how to do it?	Yes. The regulatory process for licensing of all NPPs follow essentially the same approach. The review process may involve minor differences, depending on the complexity of the design and for NPPs of standardised (repeat) design the review may not be as detailed as a new design and the inputs from the review of similar design NPPs undertaken by AERB in the past would be utilised. In such cases, the detailed reviews would be primarily focusing on the differences in design, construction, if any, and the site related aspects, unless otherwise considered necessary.	
143.	Germany	Article 14.1	Regarding the assessment of a NPP with a PSR the IAEA gives in SSG-25 Para. 2.13 a list of 14 safety factors that are	The key principle of PSR is regular and systematic review of NPP safety in comparison with current requirements / practices to identify strengths and opportunities for safety enhancements.	

			recommended to be part of a PSR. The Indian report gives a description about the topics that are part of a PSR but the Safety Factors Organisation, Emergency planning and Radiological impact on the environment are not mentioned. Could India please elaborate further how the regulatory body ensures that these topics are reviewed regularly?	The review approaches specified in AERB/SG/O-12 are consistent with IAEA/SSG-25. The safety factors considered in the PSRs of Indian NPPs are in line with the SSG-25 The safety factors that are evaluated during PSR in India also cover organisation and administration, emergency planning and environmental impact.	
144.	Peru	Article 14.1	In the enforcement section, an example is given about an accident during construction of RAPS-7&8 resulting a worker deadly injured and AERB suspended the operation. This accident may be considered as an industrial accident. Does AERB regulate and control this kind of situations or another organization is empowered for it?	Yes, the legislative framework established in India has authorized AERB to administer the provisions for ensuring industrial and fire safety aspects in units of DAE. Further details are provided in 1st para on page 33, section 7.2.3.1 and section 8.1.1 of the Indian National Report.	
145.	Russian Federation	Article 14.1	Could you please give PSA results (quantitatve risk assessments).	PSA results meet the specified targets.	
146.	Russian Federation	Article 14.1	Para 14.1.1 of the Report states that periodic safety reviews of Indian NPPs are carried out after five years of operation and the subsequent PSRs of these NPPs are carried out at 10 year intervals. It	Kindly refer to the answer for question no. – 139 posed by Australia under Article 14.1.	

			also mentions reports on periodic safety assessment developed every five and ten years. What are the differences of these safety reports?		
147.	United Kingdom	Article 14.1	The safety reviews during the consenting process do not appear to consider obsolescence of Instrumentation and Control (I&C) components and equipment. For example page 102 identifies a number of items that the Atomic Energy Regulatory Board (AERB) requires the utility to establish at the point of the commissioning safety review - this does not include obsolescence of C&I components and equipment. Please provide further information on: • India's I&C obsolescence management process throughout the design, manufacture & procurement, installation and commissioning lifecycle, • how this aligns with IAEA guidance SSG-39 and NS-G-2.12 or other relevant modern standards	The safety reviews during the consenting process do consider obsolescence of Instrumentation and Control (I&C) components and equipment. This aspect is a part of the Programmes for Ageing Management mentioned in page 102 (point iii). The AERB safety guide on life management AERB/SG/O-14 covers the aspects related to Management of Ageing of Instrument and Control Equipment and considers obsolescence as potential to cause maintainability/ operability problems in I& C systems leading to their deterioration before the end of the plant life. To overcome this it requires NPPs to have an ageing management strategy for the I&C systems. AERB Safety Guide AERB/SG-07 on Maintenance of NPPs require that a minimum number of spares to be available for components and equipment of I&C systems. Generally obsolescence related issues are identified and solutions are devised by the utilities. The regulatory reviews carried out during the design, commissioning and operation life cycle of the I&C system consider the requirements spelt out in the	

				IAEA requirement document /guidelines, in addition to the AERB codes and guides.	
148.	United Kingdom	Article 14.1	The utility prepares its periodic safety reviews (PSR) in accordance with AERB Safety Guide AERB/SGO/O-12. Only very limited details of the content of this guide or the process are provided. Please explain how consistency with IAEA Safety Guide SSG-25 has been ensured and in particular how the 14 safety factors in the IAEA safety guide have been addressed in PSRs.	Kindly see the answer to question no 142 posed by Germany under Article 14.1.	
149.	United Kingdom	Article 14.1	The national report states that periodic safety reviews are used as the basis for licence renewal and three have been completed since the last review at Tarapur (TAPS-1&2), Karapur (KAPS-1&2) and Madras (MAPS-1&2). International expectation is that the licence renewal process should ensure that any essential improvements are completed before the new licence is issued. Please provide some examples of improvements that AERB have required to be completed before licences were renewed for TAPS-1&2, KAPS-1&2 and MAPS-1&2.	India is perhaps the only country which has a system of 5 yearly renewal of license for operation of NPPs and ten yearly PSR forms one of the basis for renewal. This should not be mistaken with the practice in some of the countries where PSRs are mandated for consideration of Long Term Operation (LTO), which is essentially for allowing plant operation beyond the original design life. India also has clear approach for implementation of safety enhancements at the existing NPPs, following the international best practices. As per license renewal process, the renewals are given only on the basis that the plants meet the specified acceptance criteria. If any NPP is not found fulfilling the acceptance criteria, their license cannot be renewed. There has been no case of the NPPs not meeting the acceptance criteria as part of the licence renewal.	

				Other safety enhancements, which are aimed at further improving the safety of NPP, are prioritised based on their safety significance, need for development of solutions, detailed design, planning, procurement and opportunity for implementation. These plans and schedules for implementation are reviewed by AERB prior to renewing the license for operation PSR process. Post the license renewal, AERB monitors the progress of implementation of identified measures by the NPP, as part of the continuous monitoring of the NPP. There could be cases of conditional extensions, to facilitate planning and implementation of identified improvements, as per the agreed programme. The important safety enhancements identified for follow up of implementation in the context of renewal of license for TAPS- 1&2, KAPS – 1&2 and MAPS – 1&2 relate to the implementation of long term actions identified as part of the post- Fukushima enhancements.	
150.	United Kingdom	Article 14.1	The national report states that renewal of licenses is based on a comprehensive safety review once in 5 years and a periodic safety review, once in 10 years. There are only limited details of the scope of either review in the report. Please explain the difference in the scope of the two types of review	Kindly refer the answer to question no 139 posed by Australia under Article 14.1.	

151.	France	Article 14.2	Concerning nuclear power plants in operation, what is the status of implementation of all the challenges identified in the report of the rapporteur in Country group session in India presentation of the CNS 2014?	The Rapporteur's Report on the country group session on Indian National Presentation identified three challenges. They were (a) implementation of containment filtered venting system, (b) implementation of measures for hydrogen mitigation and (c) Readiness for review of new reactor designs. The status on implementation of these were included in the Summary of the National Report under title "Challenges and Planned Measures" as well as under different sections of the Report (a&b-section 6.5.1, c-section 14.1.2, 14.2.1 & 18). In short, the status is as follows. Challenges a & b forms part of the long term actions of the post-Fukushima enhancements,where development work has been completed and implementation has been initiated and the same will be completed in a phased manner over next two years. Challenge c: AERB has already brought out the Code on Design of LWR based NPPs in 2015, which is in line with the latest international standards. The revised requirements for site evaluation of nuclear facilities were also issued in 2014. The safety assessment of new NPPs is being carried out in accordance with the principles and requirements in these Codes. Also kindly see answer to questions no 3 and 129 posed by Canada under Article - General and Article - 14 respectively.	

152.	Netherlands	Article 14.2	Please elaborate on the plans to increase the on-site regulatory surveillance.	AERB is working on multiple options of increasing the on-site surveillance. These include the increased number of inspections by headquarter staff, inspections by staff at regional centres and deployment of on-site observers at some sites. The final decision in this regard will be taken up after assessing these options.	
153.	United Kingdom	Article 14.2	In the programme for continued verification of safety the maintenance programme states that, one of its functions is to ensure 'the safety status of the plant is not adversely affected due to ageing, deterioration, degradation or defects of plant'. In addition, it states that the programme related to life management is used to obtain information on behaviour of the Structures Systems and Components (SSCs), as identified for ageing management purposes. Please provide further information on the output of the life management programme as follows; • How is this used to inform the Instrumentation and Control (I&C) ageing management process and in turn how does it influence In-	During service, the condition and performance of these components is checked regularly as per the surveillance requirements which include continuous monitoring, instrument check, functional test, calibration and response time check as prescribed in the Technical Specifications for operation of the NPP. Moreover a preventive maintenance program is established at all NPPs covering I&C systems, to detect any degradation in the components. Based on the assessment, corrective actions such as adjustment, repair or replacement of these components is done, as appropriate. The feedback of this maintenance program is considered in the life management program of these components also. This approach is in-line with Section 2.16 of IAEA Safety Guide "Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants (NS-G-2.6)". In developing the ageing management, maintenance and in-service inspection programmes of I&C components of NPPs guidance available from the following have been used. i. AERB Safety Guide on Maintenance of NPPs	

			Service inspection to identify ageing effects and ageing mechanisms? • How this aligns with the IAEA guidance on Maintenance, Surveillance and In-service Inspection (NS-G-2.6) in respect of I&C activities? • Advise what standard(s) or guidance are used to identify ageing of I&C components and equipment during In-Service inspections or maintenance?	(AERB/SG/O-7) ii. AERB Safety Guide on Surveillance of Items Important to Safety in Nuclear Power Plants' (AERB/SG/O-8).” iii. AERB Safety Guide on Life Management of Nuclear Power Plants (AERB/SG/O-14). iv. IAEA TECDOC on Safety Aspects of Nuclear Power Plant Ageing (IAEA-TECDOC-540) v. IAEA TECDOC on Management of Ageing in I&C equipment in Nuclear Power Plant (IAEA-TECDOC-1147) vi. IAEA Safety Report on Implementation and Review of a Nuclear Power Plant Ageing Management Programme (IAEA Safety Report Series No. 15) vii. IAEA Safety Guide on Ageing Management of NPPs (IAEA/NS-G-2.12) viii. IAEA Safety Report on Safe Long Term Operation of Nuclear Power Plants (IAEA Safety Report Series No. 57)IAEA Safety Guide on Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants (IAEA-NS-G-2.6).	
154.	Canada	Article 15	It is stated that AERB is in the process of collecting inputs from various stakeholders on identification of work practices having potential for eye lens exposure and their dose estimation and development of eye lens dosimeters for revising and implementing the regulatory dose	The study related to eye dose profile of occupational workers at NPPs and other facilities including medical is in progress. The inputs obtained so far indicate that there is no potential for significant exposure to eye lens in comparison with the whole body exposure in the work practices and activities encountered in the NPPs. However, studies are in progress with respect to	

			limits for eye lens. Taking into consideration stakeholder feedback to inform the regulatory approach is similar to that done in Canada. Could India provide further information on what data they have received and their intentions on adopting the ICRP recommendations with respect to dose limits for the lens of the eye?	enhanced usage of dosimeters and eye lens dose data set in medical practices. Based on these studies and feedback received from stakeholders, revised dose limits to the lens of eye can be implemented in coming years.	
155.	Canada	Article 15	An interesting discussion on collective radiation dose budgeting is presented on page 124. It appears that the regulator reviews, approves and monitors compliance against the collective dose budgets that the operator develops to ensure that the total is within budget. It further states that any upward revision of the budget requires adequate justification by NPP, review and approval by AERB. This level of review and approval by the regulator is far and above the traditional approach used in Canada in which the regulator monitors the licensees' processes for planning, tracking and reporting on collective dose for large projects.	Review and approval of collective dose budget by SARCOP (Safety Review Committee for Operating Plants) of AERB is with the aim of continual efforts for reduction of collective dose in NPPs as an ALARA measure. This practice was introduced more than two decades back and has proven to be effective method for minimizing collective dose without compromising the operation and safety. Annual collective dose budget is prepared considering various planned activities during the year and takes account of long term program for continual improvement involving engineering and policy measures and procedural changes. Utility is having enormous information on the experience of dose consumption in different type of activities, which makes them capable of preparing a realistic budget. Revisions are necessitated in the budget primarily due	

			Is this type of review and approval in India mandated by regulations? Would India comment on how challenging and/or conservative the dose budgeting process is considering that the licensee requires regulatory approval to make changes?	to unforeseen activities/developments. In such cases, revisions are accepted and approved based on justification and the experience from such cases is utilized as part of operating experience.	
156.	France	Article 15	Has India examined the opportunity to extend the concept of dose constraints prescribed for temporary workers to all the workers?	The concept of dose constraint is prescribed for the regular as well as temporary worker. Dose constraint for regular workers is 20 mSv/year. The actual average annual dose to the monitored NPP workers is 1.15 mSv. No radiation worker received radiation dose above 20 mSv/year in the last three years.	
157.	France	Article 15	The report outlines the concept of discharge constraints. Which entity is in charge to set this constraint for each NPP?	Discharge constraints are set by utility and approved by AERB.	
158.	Germany	Article 15	India reports on the radiological protection of the public but no statements regarding the conditions for the release of radioactive material to the environment, operational control measures and main results are made. Could India please provide this information?	In India, disposal of radioactive effluents from nuclear facilities is governed by The Atomic Energy (Safe Disposal of Radioactive Wastes) Rules, 1987. Limits and conditions for radioactivity discharges to the environment during operation of the NPPs are included in the Technical Specification for the Operation of the NPPs. The limits are based on the dose apportionment for the individual facility at the site, design and site related aspects of the facility, past	

				experience related to effluent generation, management and releases from the facility or similar facilities, and application of ALARA approach. The dose to members of public on account of the discharges is assessed through environmental surveillance. It has been seen that dose to the members of public has remained at very small percentage of the specified limit of 1000 μSv/year. Public dose due to release of these radioactive effluent from different NPP sites were in the range of 0.001 - 41.01 μSv/year (0.01%-4.1% of the annual limit).	
159.	Slovenia	Article 15	RADAS readings … in plant control room and in the shift physicist's Office Q.: Are the readings of the RADAS system available also off-site and where?	RADAS readings are available in Control Room and shift Health Physicist's Room. RADAS readings are not available off-site.	
160.	Switzerland	Article 15	Only the average annual dose of the monitored persons during 2013 - 2015 is given. Due to the fact that different types of NPP are under operation in India, this comparison would give interesting information. Could you please provide more information on these issues: • A comparison of the collective doses accumulated in each Indian NPP during the last 10 years is missing and should be presented in a diagram. • The development of the average	Considering the range of values is similar to those of previous National Report 2014 (for the year 2010 to 2013), average values are provided in the current National Report. The details of collective doses in each Indian NPP were reported in earlier National Report (2014), with the comparison between old NPP and new NPP. The details of collective dose accumulated in each NPP and average individual dose in each NPP is also reported in AERB annual report which is available on AERB website (www.aerb.gov.in). The summary of these results are given in the current National Report.	

			individual dose over the last 10 years in each NPP should be given in a figure. • The distribution of the individual doses of the NPP staff and the temporary workers is to be plotted over the last 10 years.	(Please refer section 6.1.1 of National Report 2014 report, Figures 6.1 & 6.2 for collective doses consumed for older and new plants respectively).	
161.	Switzerland	Article 15	In this chapter "exposure control and implementation of ALARA" a list is given with different actions. One says e) Minimising the internal exposure by source control. In Switzerland this approach means avoiding internal exposure. Could you please provide more information on the doses accumulated by internal exposure of NPP staff working in a BWR and a PHWR, respectively.	You are right. Internal exposure is avoidable in BWR & PWR and is nil in Indian plants. For PHWRs, internal exposure other than tritium is nil, however internal exposure due to tritium uptake adds to collective dose consumed.	
162.	Ukraine	Article 15	What technologies are used for conditioning of liquid radioactive waste, in particular evaporation bottoms, at VVER NPPs?	The technologies used for conditioning of liquid radioactive waste, in particular evaporation bottoms, at VVER NPP involve fixation in a cement matrix.	
163.	Ukraine	Article 15	Does the legislation of India envisage that radioactive waste is to be disposed by only specialized radioactive waste management enterprises?	Disposal/ transfer of radioactive waste in India is governed by the Atomic Energy (Safe Disposal of Radioactive Wastes) Rules, 1987. As per these rules, no person shall dispose of radioactive waste unless he has obtained an authorisation from the competent authority (AERB) under these Rules. The AERB has	

				specified the requirements with respect to waste disposal facilities in safety code for waste disposal AERB/NF/SC/RW Disposal of radioactive waste in India are carried out by experienced and specialised government agencies.	
164.	Ukraine	Article 15	Do NPP designs in India include institutional systems for automated environmental radiation monitoring in the NPP observation areas? Is the Indian environmental radiation monitoring network (IERMON) completely self-contained and fully independent from the NPP information systems?	Yes, NPP designs in India include institutional systems for automated environmental radiation monitoring in the NPP observation areas. Yes, Indian environmental radiation monitoring network (IERMON) is completely self-contained and fully independent from the NPP information systems.	
165.	Canada	Article 16	Are there any provisions for potassium iodide pill distribution to members of the public who could be affected should there be an accidental release?	Provisions are available for potassium iodate pill distribution to members of the public if the situation warrants so during an off-site emergency. Please refer section 16.2.5.3 (v) of the National Report.	
166.	Canada	Article 16	It states that "for multi-unit site the plant/site/offsite emergency plans are revised before issuing construction consent to a new facility". What are the criteria used to determine the regulatory requirements for appropriate plant/site/offsite emergency plans?	The emergency plans of the existing facility are revised to augment the infrastructure required and other specific precautions in view of additional construction workers at the site. However, before the fuel loading, the emergency plan including of the new facility taking account of overall layout of units, additional infrastructure, emergency assembly transport etc. is reviewed.	
167.	Switzerland	Article 16	About diverse communication systems the report states that all	In case of TSBO (total Station Black Out), provisions are available in the form of extended battery based	

			mentioned systems are available for use at all times. How will communication be ensured in case of an TSBO and/or natural disater, e.g. earthquakes with largely destroyed infrastructure ? Are all emergency response key actors equipped with satellite communication means? Ground stations possibly used for satellite communications may be damaged and unavailable as a result of the earthquake. What are the requirements on fall-back communication means with regards to the transmittal of information, data and voice ?	back up and portable chargers which can provide power to various communication means. Stations are equipped with multiple and diverse communication systems including satellite and radio based communications systems. These are also at multiple locations at each site where satellite communication systems are available (like Plant Emergency Control Centre, Site Emergency Control Centre, Off-site Emergency Control Centre) which will enable prompt communication. Further in case of extreme situation mobile vehicle-mounted satellite communication will be arranged.	
168.	Switzerland	Article 16	Concerning the On-Site Emergency Support Centre, with what kind of technology or sytem is air quality ensured, i.e. prevention of an enrichment from air with carbon dioxide in situations when personnel is forced to stay inside due to the radiological situation on-site?	Presently the main control room is fitted with survival ventilation system and fresh air supply system as necessary. A centralized On -Site Emergency Support Centre is planned at each site which will be provided with a different means of fresh air supply at different time period for the breathing requirements of personnel present in the building during post accidental scenario. The provision will be made for two conditions i.e.: i. During the initial phase of a radiological emergency, high contamination in the air is expected. At this time, the fresh air supply is not taken from the active environment through survival ventilation system, rather it will be provided through the breathing air	

				cylinders for fresh air supply to various rooms of the operating floor of the building. ii. During the later phase of an emergency, the activity is expected to reduce to low levels. Beyond this period the fresh air supply would be made through the survival ventilation system. Survival ventilation system is fitted with pre filters and combined HEPA & charcoal filters.	
169.	Croatia	Article 16.1	Decision Support Systems (DSS) are mentioned on page 136 and the practice at certain facilities is described on page 139. Are DSS tools used also in other facilities and on higher levels (district, state, national)? Are such tools checked and approved by AERB?	The decision support for emergency management and estimation of projected dose is available at all NPP sites. Two indigenously developed automated Decision Support System (DSS) are operational on experimental basis at two NPP sites. DSS for NPPs are reviewed and accepted by AERB. These automated DSS are being implemented at all sites based on the experience and field testing. Also, country wide radiation monitoring network is available (IERMON) for facilitating decision making at various levels (district/state/national).	
170.	Croatia	Article 16.1	Administration of prophylactics is listed as one of the protective measures which could be taken to mitigate the consequences of an accident. Could you briefly describe the general strategy for the implementation of this measure? Have the tablets been predistributed among the population living in the vicinity of the nuclear facilities?	Sufficient quantity of prophylactics is maintained at recognized centres like Off-site Emergency Control Centre, health centres, hospitals etc. The distribution of prophylactics is to be done during emergency based on established protection strategy governed by respective predefined EALs/OILs. Pre-distribution of prophylactics among the population living in the vicinity of the nuclear facilities is not done.	

171.	Russian Federation	Article 16.1	Could you please present information about key results of the exercises conducted and their lessons learned.	Continual improvement was observed in the harmonized working of various participating agencies during the emergency exercises over the year. It was observed that response time of a few activities was at times more than envisaged but it has improved with more number of emergency exercises conducted. One of the lessons learnt was that effectiveness of preparedness should also be checked by conducting the exercise at odd hours.	
172.	Canada	Article 16.2	What are the criteria used to determine precautionary action zone (PAZ) and urgent protective action planning zone (UPZ) boundary distances?	At preparedness stage, the criteria used to determine precautionary action zone (PAZ) and urgent protective action planning zone (UPZ) boundary distances are based on hazard analysis (for all facilities in a site) carried out for wide range of accident scenarios (design basis accident, design extension condition without core melt down and design extension condition with core melt down) to meet the requirement of protective actions during emergency. In practice identical distances for these zones have been specified for all plants. During an actual emergency situation, for implementation of specific protective measures, the size of PAZ and UPZ will vary based on observed EALs/OILs during emergency. The criteria to determine PAZ are based on the prevailing emergency conditions at the facility and also on meteorological conditions. UPZ boundary distances are based on environmental monitoring or, as appropriate, prevailing conditions at the facility.	
173.	Croatia	Article 16.2	Is there any cooperation between India and neighboring countries in	Neighbouring countries are at large distances from the location of Indian NPPs. No trans- boundary	

			the field of emergency preparedness (information exchange and assistance on bilateral basis, organizing exercises, coordinating the response etc.)?	implications are expected. India being a contracting party to ‘Convention on early notification of a nuclear accident’ will notify to IAEA in case of any accident at Indian NPP. India also participates in ConvEx exercises conducted by IAEA.	
174.	Croatia	Article 16.2	On page 127 it is explained that "...for the purpose of emergency preparedness, sizes of the zones & distances are based on hazard analysis ...". Does that mean that the sizes of the emergency planning zones (PAZ, UPZ, EPD and ICPD) differ for each facility?	Kindly see the answer to question no 171 posed by Canada under Article 16.2.	
175.	France	Article 16.2	How the protective actions decided with dose projection is coordinated with the emergency classifications and especially with the EALs?	EALs for emergency classifications are predefined based on the consequence analysis for specific plant. Protective Actions are to be taken based on projected dose for the EALs which could result into radioactive releases.	
176.	Ireland	Article 16.2	It is noted that ‘EPR plans cover all emergency situations envisaged so that a graded response consistent with the gravity of the situation can be ensured’. Can India provide an example of a graded response in an emergency situation?	EPR plans are designed to cover all emergency situations envisaged so that a graded response consistent with the gravity of the situation can be ensured. This can be demonstrated in following ways: - Classification of emergency (Emergency alert, plant, site & offsite emergency). - Graded approach for protective actions based on projected dose so that it does more good than harm. Based on detailed analysis of the emergency scenario,	

				plant specific emergency action levels are defined for various situations which can lead to an emergency situation such as Emergency alert, plant, site & offsite emergency. Graded response based on EALs/OILs is implemented through various measures like defining emergency planning zones (PAZ, UPZ & LPZ), classification of response actions (Precautionary Urgent Protective Actions, Urgent protective actions, Long term protective actions). Based on projected dose, emergency response actions like sheltering, prophylaxis administration, evacuation and control of local produce etc. are well defined.	
177.	Ireland	Article 16.2	The report notes that generic criteria of greater than 100mSv/y is used for justified protective actions and 20-100 mSv/y is used for optimization of protective actions. Are the specified dose levels applicable to members of the public or emergency workers?	These dose criteria are applicable to members of the public.	
178.	Ireland	Article 16.2	Can the AERB provide some examples of the plant parameters and conditions that are used as Emergency Action Levels (EALs)?	The examples of Plant Parameters used as EAL based on impairment of critical functions are: • Sub criticality: Neutronic Signals high, during Shut down condition failure to maintain reactor in long term subcritical state due to decrease in poison concentration in moderator system • Core Cooling: Primary Heat Transport (PHT) system Pressure low , Calandria Level Low during Design Extension Condition i.e. failure of cooling through PHT system , complete failure of ECCS and failure to	

				injection to PHT system from various hook ups. • Confinement: High Containment pressure and temperature, Hydrogen concentration in the containment. Some examples of plant conditions that are used as EAL are : Loss of coolant accident with complete failure of ECCS including recirculation, Flood, tsunami or cyclone exceeding design basis, status of containment, etc.	
179.	Ireland	Article 16.2	Where is the backup control room located (offsite or onsite)?	Backup control room for plant operation and control is located onsite.	
180.	Ireland	Article 16.2	Has India tested the arrangements for evacuation in emergency exercises?	Yes, arrangements of evacuation are tested during emergency exercises. During site emergency exercises, evacuation of non-essential staff from the site is planned and tested. During offsite emergency exercises, evacuation of public on sample basis in affected area is planned and tested.	
181.	Ireland	Article 16.2	What, if any, plant parameters are available to staff in the Off-Site Emergency Control Centre?	In the Offsite Emergency Control Centre, plant parameters are not available online, but well established communication exists between plant control room & offsite emergency control centre. Plant parameters, radiological status, meteorological parameters, effluent release data and other information are continuously communicated from plant to Offsite Emergency Control Centre through dedicated communication lines.	
182.	Ireland	Article 16.2	It is noted that TLDs will be used during an emergency situation. Can	It may please be noted that both TLD as well as Direct Reading Dosimeter (DRD) will be used during an	

			India explain the decision to use TLDs (rather than electronic dosimeters, for example)?	emergency. DRD is a type of electronic dosimeter which will be used for dosimetry purposes.	
183.	Ireland	Article 16.2	Are the Environment Survey Laboratories accredited to international standards?	Kindly see the answer to question no. 59 posed by Slovenia, under Article - 6.	
184.	Sri Lanka	Article 16.2	Your report indicated that neighboring countries are at large distances from Indian power plants and no trans-boundary implications are expected. Can you indicate large distance mentioned in your report in approximate kilometers as we have experience that Chernobyl affected hundreds of kilometers.	The design of NPPs in India incorporates defence in depth which includes various safety features with the objective to prevent accidents and mitigate the consequences, should an accident occur. The neighbouring countries are few hundred kilometres away from Indian NPPs where the effect of radiation in case of accident condition is not expected.	
185.	Sri Lanka	Article 16.2	Article 16(2) of the Convention requires that each contracting party take appropriate steps to ensure that competent states in the vicinity of the nuclear power plants are provided with appropriate information for emergency planning and response. Have your Country made an assessment which states in the vicinity of power plants can affect in the case of a highest accident (INES scale 7) if occurred in a	Please refer the answer to question no. 183 posed by Sri Lanka under Article 16.2.	

			nuclear power plant of India. Would you also indicate actions taken by you to provide information for emergency planning and response to competent authority of neighboring countries that are likely to be affected by an nuclear accident.		
186.	United Kingdom	Article 16.2	The 7th Convention Report addresses the second of the five challenges arising from the 6th Convention, which focuses on achieving harmonised emergency plans and response measures. In Section 16.2.7, the report outlines how the emergency arrangements are in line with IAEA safety documents (GSR part 7, GSG-2.1, GSG2 and GSR part 3), which ensure harmonisation with international standards. For the purposes of harmonising the emergency arrangements at an international level, please clarify if India participates in any international emergency preparedness bodies (e.g. IAEA working groups) or undertakes any international emergency preparedness benchmarking exercises with other countries.	Kindly note that the National Report has addressed as to what India is doing to address all the challenges identified during the 6th review meeting of the Convention (kindly see India's answer to the question no 24 posed by Switzerland under Article – General. In specific reference to the question, India participates in IAEA technical meetings on emergency preparedness and response standards, IAEA ConvEx exercises periodically. India has also participated in IAEA IRRS mission which ensured/helped to harmonize Indian EPR plans with international standards. EPR functional areas at NPPs are also reviewed during WANO peer review as a part of international emergency preparedness benchmarking.	

187.	Canada	Article 17	In consideration of the potential impact of flooding on the NPP, is the potential for bio-fouling of the cooling water intake taken into consideration?	We presume that bio- fouling as referred here is choking/blockage of the intake due to floating bio-matter. The equipment related to safety-related cooling water system and emergency make-up provisions are located above the postulated flood level, which also addresses effect of choking of intake due to bio-fouling. Further all NPPs have on-site storage of make-up water for the important systems for ensuring safe shutdown and decay heat removal for a minimum period of seven days.	
188.	Switzerland	Article 17	It is stated that the site is assessed for floooding potential due to natural causes such as run-off from precipitation. What is the basis of the assessment of flooding potential due to run-off from precipitation: measured flow /flood height data, measured precipitation data for a specified heavy rain duration converted to flooding data via hydrological modelling of run-off or other?	Guidelines on flood hazard assessment at NPP sites are given in AERB guides, AERB/SG/6-A, “Design Basis Flood For Nuclear Power Plants On Inland Sites” and AERB/SG/6-B, “Design Basis Floods for Nuclear Power Plants at Coastal Sites”. For flooding potential due to run-off from precipitation, generally methodology based on convolution of heavy rainfall/storm via hydrological modelling is adopted.	
189.	Switzerland	Article 17	It is stated that flood waves caused by failure of upstream dams / barrages is assessed with respect to the safety of the installation. What are the characteristics of the dam / barrage failure (e. g. partial or complete failure, instantaneous or progressive break) assessed?	As a practice based on observed data of past failures of dams, following guidance is provided in AERB/SG/6-A, “Design Basis Flood For Nuclear Power Plants On Inland Sites” : “ • In case of rock or earth filled dams, the failure is not instantaneous and it develops slowly. Periods for total failures can be as large as several hours also.	

				• Arch dam failure due to flooding is likely to be instantaneous and the destruction is complete. In case non-failure cannot be demonstrated then total failure is to be considered • Concrete gravity dams are to be analysed for overturning and sliding. Size of breached section and its location should be computed consistent with the type of dam and other relevant parameters. If not, the opening shape and size of failure should be limited by a rectangular shape with the full height as one side and the bottom width of the dam structure as the other side.”	
190.	Switzerland	Article 17	It is explained that a large volume of seismological data is collected during site investigations, however there is little detail on how it is employed. Is a formalized approach for the calculations and expert judgement required in deriving a hazard from those data?	Formalised approach exists for the calculation/derivation of hazard, which includes national level expert elicitation. All these are inputs to the derivation of seismic hazards and regulatory decisions.	
191.	Switzerland	Article 17	What measures would be acceptable to engineer a safety-relevant building against liquefaction?	Susceptibility to liquefaction has to be assessed during the siting stage of an NPP. Unless engineering solutions are demonstrated to be available, site is rejected. Guidelines for the assessment and possible ground improvement techniques are covered in AERB safety guide “Geotechnical Aspects And Safety Of Foundation For Buildings And Structures Important To Safety Of Nuclear Power Plants” (AERB/NPP /SG/CSE-2, 2008).	

192.	Switzerland	Article 17	It is stated that for each of the natural and man-made hazards, whose potential at the given site is known to exist, a design basis event is established. How are design basis parameters for extreme meteorological and man-made hazards established?	Methodologies for establishment of design basis for meteorological parameters are given in AERB guide AERB/NF/SG/S-3, "Extreme Values of Meteorological Parameters". Similar guidance with respect to human induced events are covered in AERB guide AERB/NPP/SG/S-7, "Evaluation Of Design Basis For External Human-Induced Events For Nuclear Power Plants".	
193.	United Kingdom	Article 17	The report refers to "exclusion zones" and "natural growth zone(s)" around nuclear power plant sites. Please explain / clarify: • What criteria are used to determine the physical size of these zones for each category of facility? • What limits or restrictions apply to control the influx of population within natural growth zones and which agency or organisation is responsible for applying them?	(a) As per AERB Safety code on Site Evaluation of Nuclear Facilities (AERB/SC/S rev.1); the physical size of exclusion zone (EZ) is based on the following: " (i) The size of the exclusion zone around a nuclear facility shall be such that : (a) During normal operation, prescribed dose limits shall be met at EZ boundary considering all radiation exposure pathways including inhalation and ingestion routes. (b) During governing design basis accident (DBA) conditions, acceptable dose limits shall be met at EZ boundary considering all radiation exposure pathways including inhalation and ingestion and without taking any credit for emergency countermeasures in public domain. (iii) In case of NPP, the size of EZ shall not be less than 1.0 km from the center of each reactor. (iv) The size of EZ shall also satisfy the requirements with regard to security considerations of the facility." (b) Natural growth zone is established by	

				administrative measures where only natural growth is permitted. These administrative measures are applied by respective state governments. This zone is synonymous to precautionary action zone of emergency planning.	
194.	United Kingdom	Article 17	The report states that "the regulatory system also incorporates a system of 'special safety reviews', undertaken following major events / developments, wherein the implications of such experience and lessons are reviewed for identifying and implementing safety enhancements". Please clarify / identify: • What type of event or development initiates a special safety review? • Which agency or organisation is responsible for: a) Deciding that a special safety review is required and the scope. b) Identifying and implementing safety enhancements.	As of now there is no formal procedure or criteria for initiating the special safety reviews. The special safety reviews are in addition to the well-established processes of operating experience feedback, continual safety reviews and the periodic safety reviews. As practiced so far, the events / developments / new findings, etc. having significant or generic concern for safety or significant potential for safety improvements / lessons are selected. The examples could include major incidents, international or domestic, findings from inspections, safety reviews or research, for Indian plants or findings from the safety reviews done elsewhere could initiate such special safety reviews. A few examples of past instances of such special reviews undertaken for Indian NPPs and the resulting improvements are listed in Section 6.5 of the National Report. These include the Three Mile Island accident of 1979, the Chernobyl accident of 1986, the fire incident at Narora Atomic Power Station (NAPS) in 1993, the flood incident at the Kakrapar Atomic Power Station (KAPS) in 1994, the tsunami at the Madras Atomic Power Station (MAPS) in 2004, the Fukushima accident in 2011, and the pressure tube leaks at KAPS in 2015-16. There have been numerous other	

				examples including review of IGSCC vulnerabilities, 1983 incident of pressure tube failure in Pickering NGS, the Bhuj earthquake of 2001, thinning of elbows in PHT system feeders in CANDU reactors, etc. Such reviews are generally initiated by AERB in order to learn from the event and consolidate the outcome in form of regulatory requirements / guidance as appropriate. However, safety being prime responsibility of the licensee, utility may also decide to carry out such reviews. The reviews could also be carried out independently by the utility and the regulatory body. Generally, identification of the safety enhancements is by the utility. However, if the reviews by the regulatory body bring out the need for additional enhancements or need for reinforcing the requirements, they are also considered for implementation at NPPs.	
195.	France	Article 17.1	India points out that credible combination of hazards are considered. Could India give the list of combinations usually considered?	Certain guidance in respect of combinations of hazards are specified in AERB guide AERB/SG/S6-A, “Design Basis Flood For Nuclear Power Plants On Inland Sites” Some of these include Dam failure caused by an earthquake equivalent to SSE coincident with peak of 25 years flood; Inadvertent opening of all gates on an upstream dam coincident with peak of flood caused by one half probable maximum precipitation (PMP), etc.	

				The potential for internal hazards such as flooding, missile generation, pipe whip, jet impingement, and fluid release from failed systems or other plant on the site is taken into account in the design of the plant. Some external events may initiate internal fires or floods and may cause the generation of missiles. Such interaction of external and internal events is also considered in the design, wherever appropriate. While conducting safety assessment post Fukushima, it was also brought out that for inland sites, scenario involving combination of flood due to dam break and earthquake should be considered whereas NPPs along Indian coast would only be subjected to either a local earthquake or a tsunami caused by a far away earthquake.	
196.	Netherlands	Article 17.1	The Vienna Declaration for new plants requires practical elimination of any early or large release. It might not be a good approach to comply with the declaration using a cutoff frequency (the value 10x E-7). Please elaborate?	The quoted text from the National Report is not in the context of practical elimination of early or large release as identified in the Vienna Declaration on Nuclear Safety. The cut off frequency as mentioned in the text is referred in the context of screening-in of various external events / phenomena that needs to be considered for detailed assessment and establishment of design basis of particular event at a given site.	
197.	Canada	Article 18	The following statement is found in the report: “Comprehensive deterministic safety analyses and probabilistic safety assessments…” However, it is believed that	Comprehensive Level-1 PSA is carried out to identify any weak links and to achieve a balanced design in term of risk from various event sequences. Regulatory guidance on PSA including that on fire, seismic events and flood are available in AERB manual	

			insufficient detail is provided about the scope of the probabilistic safety assessment. Can the Contracting Party clarify whether a fully developed PSA is prepared for seismic events, internal fire, flooding, and high winds?	AERB/NPP&RR/SM/O-1 (2008). In line with this document, development of PSA models for seismic, internal fire and flooding events have been completed. The design of major safety related NPP structures in India are governed by seismic considerations and the loads from the high winds are less than the seismic loads, PSA for high winds is not undertaken in the current phase of analyses (Also kindly see India's answer to Question- 20 posted by Slovakia under Article - General).	
198.	Canada	Article 18	The report states that "For finalizing accident management measures, NPCIL carried out a number of analyses of postulated severe accident scenarios for ascertaining the need for installing Containment Filtered Venting System (CFVS). This study indicated that owing to design features, some PHWR units do not need CFVS, whereas requirement was considered in remaining PHWR units and TAPS-1&2." CFVS was strongly recommended in all designs by the Working Group on Analysis and Management of Accidents (WGAMA) (NEA) on Filtered Venting in June 2014.	From the accident analysis carried out for PHWRs of lower capacity and large containment volume, it is seen that the containment pressure remains within its design pressure for 7 days into the accident. This time is considered adequate to make alternate provisions for containment cooling. (Please refer Attachment titled 'Note on Features of Indian PHWRs' provided by India).	» Note on Features of Indian PHWRs

			Could NPCIL explain why specific PHWR designs were determined to be exempted from installing a CFVS?		
199.	Switzerland	Article 18	Regarding the implementation of Containment Filtered Venting Systems (CFVS), India has made substantial progress. Analyses of severe accident scenarios for ascertaining the need for installing CFVS were conducted, the design of such systems was finalized, and the detailed design is under regulatory review. The studies related to CFVS indicated that owing to certain design features, some PHWR units do not need CFVS. Could you please elaborate on such design features?	Kindly see India's response for Question no. 197 posed by Canada under Article 18.	» Note on Features of Indian PHWRs
200.	Switzerland	Article 18	The aim of the design of CFVS was to ensure contaiment depressurization during severe accidents and to achieve decontamination factor more than that considered in the radiological release assessment. Is it possible to give some information regarding the retension rate and design principals of the filters? Are there also provisions to remove organic iodine?	CFVS design of Indian PHWR is based on wet scrubbing using venturi scrubbers. Decontamination Factors (DF) have been established for large range of flows through CFVS by using air/steam through the system. Observed DFs on experimental set ups for elemental iodine, CsI aerosol and methyl iodide are much higher than the values used in radiological release assessment.	

201.	Switzerland	Article 18	There are several suppliers, e.g. Westinghouse, Areva, etc., who designed and constructed CFVS for a large number of NPPs worldwide. India decided to develop the detailed design of CFVS "in-house", at NPCIL. What were the considerations to develop an in-house solution instead of procuring an existing system?	Please refer to India's response to Question No. 63 posed by Switzerland under Article – 6.. It is upto the contracting party to decide on the mode to be adopted for implementing safety enhancements.	
202.	France	Article 18.1	India indicates that provisions are taken to limit the consequences of severe accident situation up to dose criteria specified in the table 5 of the report. Some situations may be difficult to mitigate, for instance a containment bypass, a high-pressure core melt or the melt of the fuel in the spent fuel pool. Do India request the applicant to "eliminate practically" these situations, ie to make them very unlikely with a high level of confidence?	The safety requirement of radiation dose limits for member of public due to occurrence of a 'Design Basis Accident' or a 'Design Extension Condition without core melt' has been specified in the AERB code. It is also required that design should demonstrate that in case of a Design Basis Accident, there need not be any emergency countermeasures in the public domain. In case of design extension condition without core melt, limited counter measures in terms of food control may be acceptable. In case of design extension condition with core melt, design goal remains that emergency actions will be required for limited time and area. There should not be any situation which will call for permanent relocation of members of the public. These aspects are given in Table-5 of the report. Further, the design of NPPs shall be such that design extension conditions that could lead to large or early releases of radioactivity are practically eliminated. For design extension conditions that cannot be practically eliminated, only protective measures that are limited	

				in terms of area and time shall be necessary for protection of the public, and sufficient time shall be made available to implement these measures. The design and regulatory assessment of new NPPs will be done to meet these requirements. (Refer 14.1.2.2/Page-101 of the National Report).	
203.	France	Article 18.1	India mentions some requirements of redundancy, independence, physical separation to be applied during the design of a new plant. How PSA contributes to the safety case?	Kindly see the answer to Question no 133, posed by Switzerland under Article 14.	
204.	China	Article 18.2	According to Article 18 (2) in the guidelines, “Analysis, testing and experimental methods to qualify new technologies, such as digital instrumentation and control equipment” should be included, but only the process of Independent Verification & Validation (IV&V) has been mentioned, which is not complete for adopting digital equipment in nuclear industry, especially in safety class. Question: What have been or will be done related to this issue?	AERB has specified elaborate requirements w.r.t computer based systems for systems important to safety in the Safety Code on “Design of PHWR NPPs (AERB/NPP-PHWR/SC/D (Rev-1)” and the Safety Guide on Computer Based Systems of PHWR (AERB/NPP-PHWR/SG/D-25). The requirements and guidance of these documents are similar to that of the IAEA Standards with respect to development and qualification of computer based systems for safety related applications in NPPs. The examples of the analysis and testing activities required for qualification of such computer based systems include (a) diversity and common cause failure analysis, (b) single failure analysis, (c) hardware reliability analysis, (d) environmental qualification tests such as EMI/RFI, (f) environmental cycling test, (g) harsh environment qualification (if applicable), and (h) seismic qualification tests, apart from independent verification and validation. Further, system level tests,	

				such as functional test, performance test , stress test, stability test, failure mode testing, interface testing etc. are also required for qualifying a computer based systems for use in systems important to safety. They are also required to incorporate built-in self-diagnostic features for prompt detection of malfunctions in software as well as hardware. Further as per regulatory requirements in India, computer based safety systems are to be backed up by hardwired systems.	
205.	China	Article 18.2	As the passive safety systems are employed in the design of Indian Pressurised Water Reactor (IPWR)according to the paragraph in 18.2.2. Question: What is the extent of use of passive safety features in this new design to enhance safety?	Generally, the active systems are backed with passive features. The pre-consenting review process of AERB is based on overall safety requirements. The review is in progress and quantification is not possible at this stage.	
206.	Canada	Article 19	In response to the Fukushima accident a significant safety analysis appears to have been completed to determine the required mitigating actions, subdividing actions between short-, medium- and long-term plans. However, much of the corrective actions are to be implemented as part of the long-term plan. Can the Contracting Party provide the plan timelines showing when facilities	Kindly see the answers to questions no 3, 129, 197, posed by Canada under Article - General, Article - 14 and Article - 18 respectively.	

			are to have their long-term actions completed?		
207.	Switzerland	Article 19	Did AERB ever refuse a renewal of operating licence? If so, what were the reasons?	There has not been any case so far, where AERB had to refuse any application for renewal of operating license submitted by the NPP utility. Kindly also see answer to question no 148 posed by United Kingdom under Article 14.1.	
208.	Switzerland	Article 19.2	How many deviations from the Technical Specifications are typically detected per year and station by the Technical Audit Engineer?	On an average there was one technical specification deviation per plant in last three years, which was on account of non compliance to surveillance requirement because of continued operation of the plants. These were brought out to the notice of station management in advance by the Technical Audit Engineer and prior permission for postponement of the surveillance was obtained from the Regulatory Body, which was based on the detailed safety review and assessment.	
209.	Switzerland	Article 19.2	Is the Technical Audit Engineer empowered to order measures to restore compliance with the Technical Specification? Is he empowered to order a temporary shutdown of the plant?	Yes, Technical Audit Engineer is empowered to initiate measures to restore compliance with the Technical Specification. The authority to order temporary shutdown of the plant in case of non compliance of technical specification lies with Shift Charge Engineer / Plant Management.	
210.	Romania	Article 19.4	Do the licensees perform periodic plant drills simulating the response to transients and accidents and exercising the emergency operating procedures and severe accident guidelines? If yes, what is the	Yes. The frequency of the exercise is once in a year for each operating crew. The exercises are conducted on severe accident management provisions to demonstrate their functionality in accordance to Accident Management Guidelines.	

			periodicity of such exercises and how are they conducted? Do such exercises include the simulation of actions in the installations and on site?		
211.	Romania	Article 19.4	How does the regulator review and inspect the verification and validation of emergency operating procedures and severe accident management guidelines?	The regulatory body reviews the approach to handle emergency situations as a part of review of Safety Analysis Report. The regulatory body reviewed the generic guidelines on management of severe accident at all NPPs. The availability of plant specific EOPs and SAMGs are verified during regulatory inspections. AERB has also independently verified selected analyses related to SAM. Additionally, AERB checks the aspects related to operator training related to SAM as part of operator qualification.	
212.	Russian Federation	Article 19.5	How the engineering and technical support to NPPs is provided in case of accidents.	The engineering and technical support to NPPs in case of accident have been identified in the station specific documents on accident management guidelines. In the case of accident, initial response is from NPP personnel, for which training programme exists covering accidents within and beyond design basis. Technical support to the affected station is also provided from utility design and safety analysis office, for which a control room is established. From this control room, required technical support can be provided as utility has personnel having experience in design, operation and safety analysis. In addition, the Department of Atomic Energy will provide support as required by the NPP in managing the accident.	

213.	Canada	Article 19.6	Paragraphs 2 and 3 (on p. 169) discuss provisions for Root-Cause Analysis (RCA) of unplanned events at Indian reactors. Has a root-cause analysis been completed for the leaking pressure tube in KAPS-2 (July 2015)? With respect to the similar event at KAPS-1 (March 2016), Article 14.3.2 (p. 110) acknowledges that the RCA is in progress. Has a target date been set for completion of the KAPS-1 RCA?	Root cause analysis of KAPS-2 & KAPS-1 events is in progress. The affected PT of KAPS-1 has been removed from the core and has been brought to hot cells for post irradiation examination. For further details on the event and update on the progress of investigations, kindly refer attachment titled 'Note on KAPS PT Failure'.	» Note on KAPS PT Failure
214.	France	Article 19.6	Could India present with more details the system of operating experience feedback, in particular tools and databases developed from technical exchanges with IAEA and others countries? Could India draw up a balance sheet of safety significant events occurred on NPPs for the last 3 years?	NPCIL has a comprehensive Operating Experience feedback programme which has been implemented based on the guidelines given in Head Quarter Instructions (HQI) – 0540 (R-1). A database on important operational events and action taken is maintained. The number of SERs in the last 3 years 2013, 2014 and 2015 are mentioned in the last para of page-168 (para 19.6) of the report. AERB has established a well-structured OE program for utilizing the operating and regulatory experiences gained from various internal and external sources. The objectives of this program are as follows: • To enhance and ensure nuclear & radiological safety of NPPs / Projects. • To improve regulations and NPPs' processes, practices & documentation. • To share relevant OE information within AERB,	

				national & international stake holders and public. • To enhance knowledge base and technical competence of regulatory staff. The AERB has an OE Group with members from multidisciplinary fields having vast experience & knowledge of regulatory activities and nuclear & radiological safety aspects of Nuclear Power Plants / Projects. The OE process consists of activities like collection of experience / information, screening, evaluation, review & trending, dissemination, action development & their implementation / follow-up and maintenance of records (refer Fig-5 of CNS report). AERB maintains an internal data base for storage of OE related inputs and records. The actions developed during screening and evaluation of OE inputs along with responsible agency & process for implementation are uploaded in this internal database. It is designed in a way to enhance the knowledge & regulatory insight of users and to facilitate easy retrieval of information for later use. This online database is utilized to follow-up the implementation of identified actions. This database is being utilised for the core regulatory processes (licensing, safety review, regulatory inspection, enforcement, regulatory document development, etc.) During the reporting period i.e. from 2013 to August 2016, total 131 events were reported from operating NPPs, of which two were of INES level-1, one provisionally rated at INES level-1 and remaining INES level-0.	

215.	France	Article 19.6	Concerning ageing management, can India give more details on the ageing management program? What are the type and the scope of controls? What are the first results? Are there modifications implemented deriving from the controls?	The utility has an exhaustive ageing management programme, which covers all the systems, structures and components important to safety. A master list of all such items is prepared, along with the identified degradation mechanisms and the health assessment programmes are in place for each of the items. This master list is reviewed and concurred by the regulator. The requirements of having ageing management programme are specified by the regulatory body. An assurance on the adequacy of ageing management programme is obtained by the regulator at the time of periodic safety review for renewal of operating license of NPP. The comprehensive ageing management programme of the utility classifies the SSCs into 'not replaceable', 'limited accessibility and difficult to replace due to radiation exposure and /or require long shutdown period' and 'replaceable/repairable components'. In formulating the ageing management programme, priorities were assigned based on operating experience on ageing and premature failures. While preparing new ageing management programme, the guidelines given in AERB Safety Guide AERB/NPP/SG/O-14 on Life Management of Nuclear Power Plants and the IAEA guidance documents such as NS-G-2.12 were taken into consideration. The timely detection of age related degradation and its mitigation by necessary corrective measures is ensured through review programmes which include following major elements: 1. Preventive maintenance programme	

				2. In-service Inspection programme 3. Condition monitoring of equipment 4. Surveillance testing & monitoring programme as per Technical Specification 5. Chemistry control programme 6. Operating Experience feedback programme 7. Design changes/retrofitting based on operating experience and use of modern equipment/instrument. 8. Non-routine / need-based / component specific examinations and checks.	
216.	Canada	Article 19.7	It is noted that the report provides details of the OPEX feedback system, including obtaining reports of national and international events and communicating these to NPPs in India. However, it is not until figure 5 (p. 171) that it is observed that India disseminates its OE outputs with the international nuclear community. Can you briefly describe the manner used by India to share OPEX from its NPPs with the international nuclear community?	India shares OE of Indian NPPs through various international platforms like IAEA-IRS, WANO, COG, IAEA-INES and various regulator and operator forums. The utility, NPCIL shares the operating experience with WANO by forwarding WANO Event Reports (WERs) regularly which are posted on its website. On an average NPCIL submits around 40 WERs to WANO every year. AERB shares the events of Indian NPPs in IAEA-IRS. In addition, AERB shares the operating experience through the regulators forums (VVER Regulator’s Forum, IAEA Annual Meetings of Senior Regulators of Countries Operating CANDU Type Reactors, and other multilateral and Bilateral meetings).	
217.	Russian Federation	Article 19.7	Could you please clarify what is the procedure for national experience transfer to be applied by other international organizations and regulatory authorities.	Kindly see answer to Question no 215 posed by Canada under Article 19.7.	

218.	Canada	Article 19.8	"These storage bays are designed to accommodate spent fuel accumulated during 10 reactor years of operation." Is all spent fuel transferred to dry storage after 10 years, or only if space is needed in the bay?	Spent fuel is stored in a water filled storage bay provided at each NPP. These storage bays are designed to accommodate spent fuel accumulated during 10 reactor years of operation. In addition, space is also reserved for storing one full core inventory of fuel in case of exigencies. Depending upon the requirement, spent fuel may be transferred from the spent fuel storage bay to Away From Reactor-Spent Fuel Storage facility or for reprocessing. However, a minimum cooling period of 5 years is ensured before transfer of spent fuel to any of these facilities.	
219.	Canada	Article 19.8	Can examples be provided of what waste management techniques are utilized in India? Examples may include recycling, delay and decay, long term storage, disposal, etc.	The waste management techniques for different type of waste in India are as follows: For Low level liquid waste- Filtration, ion exchange, evaporation, dilution and discharge techniques are used. For Solid waste- Volume reduction for compaction, incineration and conditioning through immobilization by polymerization & cementation techniques are used. For gaseous waste –filtration, dilution and dispersion through stack. In the waste management plants, large storage provision exists for liquid waste to achieve decay of short lived radionuclide, chemical treatment of long lived radionuclide and cementation of the sludge. The long term solid waste storage for long lived radionuclides at present is done on retrievable basis. Please also refer to section 1.3 of the Indian National report to 7th RM	

220.	Switzerland	Article 19.8	What is the management strategy for spent fuel beyond the storage at reactor and away from reactor?	Spent fuel generated from operation of nuclear reactor is considered as resource for future energy needs. A closed nuclear fuel cycle program is followed for recovery and recycle of fissile / fertile materials.	
221.	Switzerland	Article 19.8	What is the management strategy for radioactive waste management from nuclear facilities?	Please refer to answer to question no. 218 posed by Canada under Article 19.8 and Section 1.3 of the India's National Report .	
222.	Switzerland	Article 19.8	Are strategic decisions referring to questions 1 and 2 up to the individual licensee or is there a national strategy?	These are National Strategies. (This answer is in relation to the questions no. 219 and 220 posed by Switzerland under Article 19.8)	
223.	Switzerland	Article 19.8	How is the minimisation issue of art. 19, clause VIII implemented is such strategies ?	Please refer to answer to question no. 218 posed by Canada under Article 19.8.	
224.	Switzerland	Article 19.8	How do you consider requirements of subsequent waste management steps (e.g. transport, storage, disposal) in strategic decisions on prior steps (e.g. conditioning)?	The requirements related to predisposal management including pre-treatment , treatment and conditioning are specified in AERB Safety Code on Management of Radioactive Waste, AERB/NRF/SC/RW, 2007. Conditioning of radioactive waste includes operation such as immobilization and packaging. Conditioning process with compatible matrix is selected to obtain a waste product to meet acceptance criteria at subsequent steps of transport, storage and disposal. Please also refer to answer to question no. 218 posed by Canada under Article 19.8.	

Note on Features of Indian PHWRs

Attachment on Answer to CNS Questions No. 40, 62, 129, 197 and 198 posted on the India's National Report for the 7th Review Meeting of the CNS

This attachment provides supporting information to Questions No. 40, 62, 129, 197 and 198 raised on Indian National Report for the Seventh Review Meeting of the Convention. This attachment brings out strengths of PHWRs in terms of multiple modes of core cooling; which provides enhanced resistance to the design against core damage accidents; and retards accident progression (postulated with multiple failures). This attachment also brings out the scenario wherein need of containment filtered venting was considered necessary.

For accidents to proceed to core damage/melt, all the possible heat sinks must fail. In PHWRs, there are multiple and independent heat sinks, which are present during normal operation and they become an important feature in retarding accident progression.

For normal power operation, heat sink is provided through steam generator – condenser route. Steam Generators are provided feed from main and auxiliary feed pumps. In absence of motive power, procedures call for depressurizing steam generators, which allows injection of water into steam generators through lower head pumps with diesel driven engines. In addition, deaerator storage tank at higher elevation provides gravity flow of water into depressurized steam generators. While the above mentioned injection of fire water can be initiated from the control room, additional connection points are provided outside reactor building from where water can be injected into steam generators, as part of safety enhancements made following the Fukushima accident. Water through these easily approachable hook-up points can be provided through portable pumps, fire tenders, etc. PHWRs in India are provided with on-site seismically qualified water storage, adequate to facilitate decay heat removal for seven days, keeping the reactor in cold shutdown state. When this inventory is used to remove decay heat through steam generators, it can last for about one month. All these provisions provide diverse means to supply water to steam generators in case of station blackout situation.

In PHWRs, short length fuel bundles reside inside pressure tubes, which are surrounded by calandria tubes. Assembly of pressure tube and calandria tube, along with other components, is called fuel channel and a reactor core consists of about 300-400 fuel channels (depending upon power rating of the reactor). These fuel channel assemblies are submerged in heavy water moderator contained in calandria vessel. This calandria vessel is kept submerged in light water within the concrete vault, called calandria vault. In addition to coolant around the fuel, collective inventory of moderator and calandria vault water is few hundred tons; which is always present as it is required for normal operation. This inventory is helpful in slowing down the progression of accident. When coolant inventory around fuel is lost (due to either loss of coolant accident or gradual heat-up in case of loss of heat sink); decay heat can be removed by the moderator, which has its own cooling system. If moderator cooling is also considered to be unavailable, then fuel channels gradually lose this water cover outside (due to heat-up and evaporation); which can be easily replenished from outside the reactor building, as part of accident management (through hook-up points). Total loss of moderator inventory due to heat up takes few hours; and during this process, core collapse and heat-up of debris is expected, if no addition of water is done. But, due to calandria vault water being present outside calandria vessel, core debris can be retained inside calandria and decay heat can be removed. Calandria vault water has its own cooling system and only on its cooling failure it will start getting heated up and lost due to evaporation, which takes tens of hours. Like in calandria, water addition can be done in calandria vault as well, as part of accident management (from hook-up points outside reactor building). Thus, these heat sinks and their augmentation as part of enhancement of accident management, provide sufficient time to take actions to inject water from outside reactor building in these low pressure systems.

From the above description of multiple and independent means of core decay heat removal and provisions made for accident management in PHWRs, following points are derived, which provide basis for the responses to Questions No. 40, 62, 129, 197 and 198.

i) In-vessel retention of core debris is possible in PHWRs by maintaining calandria and/or calandria vault heat sinks. Replenishment of water from outside reactor building (as an accident management measure) to these systems is a credible solution in view of
   a) Slow progression of accident providing sufficient time for such accident management actions
   b) These systems being low pressure systems, simple water pumping arrangements are adequate.
ii) Containment pressure rise following accident is due to evaporation of water inventory in calandria and calandria vault. When water is added to these systems (through hook-up points), its evaporation also adds to containment pressurization. However, this is a slow process and provides time to make alternate arrangements for containment cooling. Once cooling is restored, containment pressure comes down (without using containment filtered venting).
iii) In PHWRs, having water filled calandria and calandria vault, depending upon plant design/rating, containment pressure reaches to design pressure after about 1-5 days. This time can be utilized to make containment cooling/spray systems operational and bringing down containment pressure. In 220 MWe PHWRs commissioned after the year 2000, containment volume is larger and in these units, containment pressure does not reach to the design pressure even after seven days of steaming of water into calandria/calandria vault. This time is considered adequate to make alternate arrangements for containment cooling and therefore in these NPPs, containment filtered venting was not considered necessary.

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Note on Events of Leak from Pressure Tubes in KAPS-1&2

Attachment on Answer to CNS Question 45, 47, 48, 53, 68, and 212

Background

Kakrapar Atomic Power Station Units 1&2 (KAPS-1&2) are 220 MWe Pressurized Heavy Water Reactors (PHWRs) situated in the Gujarat State in India.

KAPS-2 is the lead reactor in India with (double melted) Zr-2.5% Nb pressure tubes. This reactor had operated for 15.3 Full Power Years (FPYs) / 18 Hot Operating Years (HOYs) when leak was detected in one of its pressure tubes.

In KAPS-1, originally installed Zircaloy-2 pressure tubes were replaced with quadruple melted Zr-2.5% Nb pressure tubes during En-masse Coolant Channel Replacement (EMCCR) campaign. This reactor had operated for 4.81 FPYs / 4.87 HOYs when one of its pressure tube failed.

Event Description & Progress on Root Cause Analysis

The status of investigation activities being performed after KAPS-2 & KAPS-1 events to determine the root causes and their outcomes are brought out below in chronological sequence.

KAPS-2 Event

a. On July 1, 2015, KAPS-2 experienced an event of leak from a pressure tube while operating at power level of 203 MWe. The leak was indicated by alarm generated in Annulus Gas Monitoring System (AGMS). After the indication of leak, the reactor was manually shutdown, cooled and depressurized as per established procedure and plant technical specifications. On scrutinizing the AGMS data, it was observed that the leak incipience was captured by AGMS about 30 hours before the alarm level was reached. After the reactor shutdown, it took few days to drain and dry the AGMS system and to firmly establish Q16 as a leaky pressure tube.

   The event was categorized as small leak from primary coolant system and provisionally rated at ‘Level 0’ on INES.

   The in-situ inspection of the leaked pressure tube using BARCIS (i.e. a coolant channel inspection tool) in August 2015 indicated presence of a through wall longitudinal crack of about 24 mm length near the inboard side of cold end rolled joint. This channel was removed from reactor in September 2015. The axially cut multiple pieces from the cold end side portion (having crack) of the pressure tube were taken for hot cell examinations and failure analysis.

b. After observing a through wall crack near the rolled joint; manufacturing, installation and operation data of the leaked channel were reviewed. These records indicated no abnormality in these processes.

c. Since KAPS-2 had seen operating life of 15.3 FPYs / 18 HOYs prior to the event and the observed crack was near to the rolled joint, the failure mechanism was suspected to be similar (i.e. delayed hydride cracking) to that was experienced in CANDU reactors.

   Considering this, a thorough review of the life management program for KAPS-2 coolant channels was undertaken. Many coolant channels including their rolled joint regions were also inspected after a careful selection. No abnormality was observed.

d. The hot cell examinations on one of the axially cut pressure tube pieces confirmed the presence of a tight through wall longitudinal crack of 16 mm length (with two parallel part through wall minor cracks close to the end of the through wall crack) near to the cold end rolled joint. The hydrogen concentration around the crack and up to rolled joint location was measured. The observations are in favour of a DHC phenomenon leading to leak near the inboard side of cold end rolled joint.

e. The hot cell examinations on the pressure tube pieces also revealed presence of some local corrosion spots on the exterior surface of pressure tubes. Subsequently, the remaining part of the pressure tube (i.e. main body and hot end rolled joint region) was also taken up for hot cell examinations. This revealed an unprecedented observation of localized corrosion spots on the exterior surface of the leaked pressure tube. It was suspected that the localized corrosion of pressure tube exterior surface are secondary effect of leaking coolant and might have occurred due to prolonged exposure to steam environment following leak from pressure tube.

f. The extensive literature survey done for the presence of such corrosion spots also indicated that the localized corrosion spots (termed as nodules) are possible in Zr-2.5 Nb alloys by prolonged exposure to oxygenated water / steam environment under radiation or without radiation. The time period required for such corrosion to occur at about 300^{o}C (i.e. the outlet channel temperature in PHWR) is considerably long. Thus it was suspected that a minor leak in the pressure tube might have been present for a prolonged period and AGMS was not sensitive enough to indicate such a minor leak.

However a thorough review of the past records of AGMS indicated that the system was well maintained and responding. This review did not indicate that the pressure tube was leaking for long time. The performance evaluation of AGMS at other PHWR (similar to KAPS-2) also confirmed that the system is sufficiently sensitive and even detects a leak much lower than the system design basis. Such in-depth review of AGMS configurations at all plants nevertheless has resulted in useful suggestions for its further improvement at some plants.

g. To confirm the presence of localized corrosion problem, a neighboring channel i.e. Q-15, which shares AGMS string with the leaked Q-16 channel, was also removed in February 2016 for examinations. Similar corrosion spots were also observed on the pressure tube exterior surface of this channel. This raised concerns for other pressure tubes in the KAPS-2 reactor.

   The AGMS system configuration interconnects the annuli of certain channels in the reactor. Hence, it was understood that localized corrosion would have been extended to such interconnected channels as well.

h. Subsequently, utility submitted a proposal for removal of two more channels from KAPS-2 reactor to check the presence of localized corrosion on their pressure tube exterior surface. These two proposed channels included a channel where leaking steam could have gone to the annulus and the other channel where there was no possibility of leaking steam to be present as per the configuration of AGMS system.

   Since the outcome of this exercise was not expected to result in any meaningful conclusion, the proposal was not accepted by AERB. Utility was asked in the first week of March 2016 to urgently develop non-destructive means for detection of localized corrosion spots on exterior surface of pressure tubes and carry out in-situ inspection of a number of pressure tubes.

i. For this development, localized corrosion spots were generated on the Zr-2 spool pieces. Using these spool pieces, the BARCIS tool was tuned for detection of the corrosion spots on pressure tube exterior surface and later qualified on the leaked Q-16 pressure tube removed earlier from KAPS-2. Qualified BARCIS tool was ready by beginning of April 2016.

KAPS-1 Event

j. While the investigations of KAPS-2 event were in progress, on March 11, 2016, KAPS-1 experienced an event of pressure tube failure. Following the event, the reactor underwent automatic shutdown and all safety systems (emergency core cooling system, containment isolation system, etc.) provided in the design to deal with the event got actuated and performed as intended. A plant emergency was declared immediately after the event. The post event investigation activities identified Q-15 as the failed channel. The plant emergency was terminated after safely defueling and isolating the failed channel from primary coolant system on March 21, 2016.

After pressure tube failure event, reactor underwent automatic shutdown and all safety systems provided in the design for emergency core cooling and containment isolation functioned as intended. There was no fuel failure because of the event. The event did not result in any radiation over-exposure to plant personnel. The radioactivity releases remained within the specified limits for normal operation. During the course of plant emergency, environmental survey within the site as well as in the off-site domain up to 30 km from the plant was carried out. This confirmed that there was no increase in the background radiation levels. The event was categorized as 'Small LOCA' and provisionally rated at 'Level 1' on INES.

k. The BARCIS tool with additional capability to detect presence of corrosion nodules on pressure tube exterior surface was available now after due qualifications as mentioned above. This was first used (in April 2016) in reactor for inspection of the failed channel in KAPS-1. The inspection of the failed channel indicated presence of multiple longitudinal cracks at the hot end of the pressure tube, along with localised corrosion spots on its exterior surface. A number of channels of KAPS-2 reactor were also examined thereafter by BARCIS tool after careful selection. Localized corrosion on the pressure tube exterior surface was observed in all inspected channels.

l. On such observations, AERB ordered for expeditious inspections of the coolant channels in other operating PHWRs. These inspections were undertaken by the utility on sample basis taking into account the AGMS layout and the direction of PHT flow. The selected channels were inspected in at least one reactor out of the twin unit PHWR stations all over the country. No abnormality was observed in other operating PHWRs.

m. When the inspection data obtained from BARCIS inspection of KAPS-2 channels was carefully analyzed it was seen that it follows a pattern. The density of localized corrosion was relatively high on the pressure tube exterior surface towards annulus gas inlet end and gradually reduced towards the annulus gas outlet end and was independent of coolant hot end or cold end in the channel.

   During these in-reactor inspections by BARCIS, one pressure tube (i.e. N-06) in KAPS-2 was observed to have minor part through wall cracks on the exterior surface. This pressure tube was also removed from reactor and sent for hot cell examinations.

n. Two pressure tubes from KAPS-2 reactor, which is the lead reactor using Zr-2.5% Nb material in India, were removed earlier (in years 2005& 2012) for the purpose of material surveillance as per in-service inspection programme. As part of the investigations, these two earlier removed pressure tubes from KAPS-2 reactor were also re-examined and re-confirmed to have no localized corrosion.

o. Based on the above investigation findings, it is inferred that the problem of localized corrosion on the exterior surface of pressure tubes is specific to KAPS units. This corrosion phenomena has taken place sometime after 2012 in KAPS. Relatively higher density of corrosion nodules at the end of channel from where carbon dioxide annulus gas enters is indicative of corrosion being associated with possibly to some unlisted impurity in the annulus gas.

p. The CO_2 cylinders in use for AGMS at KAPS-1&2 after 2012 are thus suspected to carry some impurity that was not existing earlier and also not in the gas being used at other stations. The CO_2 cylinders at KAPS site and other operating PHWRs were analyzed and compared for all impurities. Also the manufacturing routes of CO_2 gas used at KAPS site and other operating PHWRs were studied. It was noted that CO_2 gas received at KAPS site after 2012 was a by-product of an industry making Mono Ethyl Glycol (Naphtha cracking) and was purified before supply to the site. While for other operating PHWRs, it continues to be from fertilizer plants or molasses plants.

q. The examinations performed on the localized corrosion spots have revealed that these are white deposits of Zirconium oxide (ZrO_2) and have lenticular shape, with their major axis always oriented towards the longitudinal direction of the pressure tube. Their length and depth are typically 1.5-2.0 mm and 100-200 µm (maximum observed depth - 300 µm) respectively. On thorough examination near the rolled joint, only a few corrosion spots near the inboard side of rolled joints were observed to have longitudinal cracks. These cracks to some extent have penetrated inside the pressure tube matrix.

r. The through wall & part through wall cracks in pressure tube Q-16 (KAPS-2) and part wall cracks in pressure tube N-06 (KAPS-2) were also observed in the longitudinal direction and near the inboard side of rolled joints.

s. The investigations are in progress to understand the localized / nodular corrosion formation mechanisms on an autoclaved pressure tube surface considering nitric acid vapour, moisture, hydrocarbons, etc. as suspected impurities in the annulus gas environment. A numbers of laboratory test set-ups for this purpose are operating at many places in India.

KAPS-1

From the observations made on KAPS-2 pressure tubes, it was understood that the region near to the rolled joint in failed pressure tube of KAPS-1 requires a thorough in-situ inspection as well as detailed examinations of the localized corrosion spots in hot cell. AERB, therefore, recommended that the failed pressure tube of KAPS-1 should be thoroughly inspected in-situ before removal from the reactor. Also, the affected portion of the failed pressure tube shall be brought to the hot cell along with the end fitting so that all evidences near the rolled joint are preserved. This required development of new tools for cutting / removal of long pressure tube section & cutting of thick section of end fitting, re-designing, approval & manufacturing of a transportation flask and some hardware changes at hot cell facility. After meeting these requirements, the affected portion of pressure tube along with end-fitting was successfully removed from reactor and transported to Mumbai on January 19, 2017 for hot cell examinations.

Lessons Learned & Corrective Actions

The investigations are still in progress.

Based on the insights gained so far from the investigation findings, following corrective measures have been taken.

- The specifications as well as quality checks of the gases used in AGMS have been strengthened in all PHWRs.
- The pressure tube exterior surface of the coolant channels in other operating PHWRs have been inspected and observed to have no localized corrosion. The inspection requirement for detection of localized corrosion has been included in the ISI program of coolant channels.

The KAPS-2 & KAPS-1 events and the latest information on their investigation findings were shared with the international nuclear community through the following.

- Annual Meeting of the Senior Regulators from the Countries Operating CANDU Type reactors in November 2015.
- Event Rating Form for KAPS-1 event posted on IAEA-INES website on March 14, 2016
- AERB Press Releases, after KAPS-1 event, on March 11, 2016, March 14, 2016, March 16, 2016, March 22, 2016 and July 1, 2016.
- Communications with CNSC, Canada following KAPS-1 event
- Bilateral Meeting with Canadian Delegates on the side-lines of the IAEA International Conference on Effective Nuclear Regulatory Systems during April 11 – 15, 2016 at IAEA Headquarters, Vienna
- IRS report on KAPS-2&1 events posted on IAEA-IRS website on October 14, 2016.
- Technical Meeting to exchange experience on recent events in NPPs and Meeting of Technical Committee of IRS National Coordinators during October 17-20, 2016 at IAEA Headquarters, Vienna
- Biennial Meeting of INES National Officers during November 21-25, 2016 at IAEA Headquarters, Vienna
- Bilateral Meeting with CNSC during IAEA General Conference in September 2016 at IAEA Headquarters, Vienna
- OECD/ NEA WGOE presented the KAPS events in Committee on Nuclear Regulatory Activities (CNRA) & Committee on the Safety of Nuclear Installation (CSNI) meetings in November 2016 & December 2016 respectively. Queries raised were answered by Indian representative.
- Annual Meeting of the Senior Regulators from the Countries Operating CANDU Type reactors in February 2017.

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