Convention on Nuclear Safety
Questions Posted To France in 2014

Q.No	Country	Article	Ref. in National Report
1	Belgium	General	Summary, chap. B, sec. 3.3.5, page 24

Question/ Comment -
ASN published the national action plan for France concerning the implementation of the recommendations resulting from the European stress tests conducted in 2011 and, more generally, all the actions decided further to these tests. The action plan is referenced (by a link to the ASN web site) and discussed in several places of the report.
The action plan provides deadlines for actions. It includes complex actions where studies are followed by implementation or deployment phases. For such actions, planning risks do exist and can lead to delays (much) longer than expected, for both the study phase (related for instance to difficulties in establishing first the main principles, then the design details of the related safety improvement, or in discussions of these topics between the Utility and the Safety authority) and the implementation or deployment phase (related for instance to searching adequate equipment providers, to reaching the adequate equipment qualification requested by the provided specifications, to construction and delivery times, or to other processes associated to implementation or deployment, like writing and validating procedures for use and maintenance, training of the operators, …).
Do the deadlines provided in the action plan rely upon analyses of such planning risks, or is it foreseen to possibly adapt deadlines in the future, in case of later risk or delay identification?

Answer The deadlines for ASN regulatory demands have been determined after technical exchanges with operators which were able to present their positions. The deadlines are due requirements.

A few ASN demands required verification of some design options by the operators for them to choose the best technological or cost effective solution.

Some of these issues have been highlighted by ASN for which the operators usually study different technological options in parallel and implement a solution that fulfils the requirement and cope with the regulatory delay.

Q.No	Country	Article	Ref. in National Report
2	Canada	General	Various

Question/ Comment Can ASN clarify the expression “ASN resolutions” used throughout the text of the Report and whether these are similar in scope as an “order” to a licensee in respect to licensing activities such as granting a licence, imposing licence conditions and related regulatory expectations.

Answer	The law gives ASN competence to take regulatory decisions to clarify the decrees and orders relating to nuclear safety and radiation protection. It is then general regulations. On the other hand, the law gives the ASN jurisdiction to take individual decisions "ASN resolutions" which, before the TSN Act, were the responsibility of ministers orders. Since the TSN Act, "ASN resolution" are similar in scope as an "order" to a licensee.		
Q.No 3	Country Canada	Article General	Ref. in National Report Page 25, Section 3.4.1, Line 7
Question/ Comment	The report states that "…extending the operating life of its reactors beyond 40 years…", can you clarify what is meant by the expression "operating life"? Does it mean pre-assessed design life of Structures, Systems and Components (SSCs), especially the passive and non-replaceable SSCs or term of the operating licence? Were these reflected/documented in the original design documentation of the plant.		
Answer	In the licence of the existing French NPPs there is no time limit for operation. However, at the design stage of the NPP, an operation period of 40 years was assumed for some SSCs, such as the reactor pressure vessel, the reactor coolant system, and for the number of transients. Those assumptions were documented in the design safety reports and are updated as needed, particularly during the ten-yearly periodic safety reviews.		
Q.No 4	Country Canada	Article General	Ref. in National Report Page 184, Section 19.6.1
Question/ Comment	Please clarify if there are regulatory requirements for licensees to provide information to the public regarding on-site accidents/incidents concurrent with its notification to ASN.		
Answer	Regular regulatory inspections are organized each year. In case of an incident, an inspection will be organized on site rapidly, in the following days. The inspection will be based on the understanding of the event and the identification of the gap with the reference documents applicable. In an emergency situation, the ASN Commission could also impose actions to the licensee through ASN resolutions. In addition, the prosecutor will be in charge of the inquiry for severe situations.		
Q.No 5	Country Canada	Article General	Ref. in National Report Page 18, 1.5 Publication of the report
Question/ Comment	ASN is commended for making the National Report of France to the 6th Review Meeting of the CNS accessible on its Website, in both French and English as well as using its Website to promote openness and transparency through the sharing of outcomes of its "stress test" and follow-up to the lessons learned from the Fukushima nuclear accident. Has there been any feedback from the public in support of these initiatives? Can you provide examples?		
Answer	Regarding the National Report of France to the 6th Review Meeting of the CSN, there has been no feedback from the public.		

Concerning specifically the follow-up of the stress tests conducted within the European framework : like all other EU Member States having NPPs, France made public a national action plan. All action plans were published and the public had the opportunity to make comments in early 2013. More information is available at : http://www.ensreg.eu/EU-Stress-Tests

Q.No	Country	Article	Ref. in National Report
6	Germany	General	Summary, p. 23 – 24

Question/ Comment
For the EDF nuclear power plants, the “hardened safety core” shall comprise an additional “bunkerised” ultimate emergency diesel generator for each reactor, a diversified emergency water supply system (see § 18.3.2.2), as well as an emergency management centre able to withstand a large-scale event affecting several facilities simultaneously (see§ 16.3.1.1).
Please give an overview of relevant options of electrical supply during SBO including all planned backfittings after Fukushima.

Answer
The French NPPs are by initial design, or following periodic safety reassessment modifications, able to cope with SBO situations (EFWS turbine driven pumps, turbine driven electric generator, dedicated Emergency Operating Procedures...)
Following the accident at Fukushima Daiichi NPP, and in addition to this design, the new bunkered ultimate EDG will automatically start to cope with SBO potentially induced by extreme situations, such as those studied in the stress tests.
As a complementary back up, the "Nuclear Rapid Action Force" (FARN) team will perform fuel supply and may substitute the ultimate EDG with mobile diesel generator sets if necessary.

Q.No	Country	Article	Ref. in National Report
7	Germany	General	Summary, p. 24

Question/ Comment
As of 2012, the gradual deployment of the “Nuclear Rapid Intervention Force” (FARN) proposed by EDF (see §16.3.1.2), a national intervention system, devoted to the licensee, comprising specialised personnel and equipment, which can take over from the personnel on a damaged site and deploy additional emergency intervention means within 24 hours after operations begin on a site, less than 12 hours after they are mobilised. This arrangement may be common to several of the licensee’s nuclear sites. The system has been partially operational (for intervention on one reactor of each of the sites) since the end of 2012 and will be fully operational in late 2015 (Gravelines – for 6 plant units);
Will each shift of the intervention force be able to cope with events at any site or will they be specifically able to deal with events at one standardized series?
Please provide further details on the structure and the organization of FARN (location, shift work, will the team members be part of regular shift teams of the reactors or newly and specifically trained personnel)?

Answer
a) Each shift of the Rapid Response Nuclear Force is able to cope with events at any site. Concerning operation area, all people of the FARN (recruited in operation departments) are trained to operate any control room of EDF fleet, provided that the installations are severely damaged, and that their mission will thus be strictly limited to a very few tasks.
b) FARN comprises a headquarter (located near Paris, about 30 people), and 4 regional services (each located on a NPP representing

one serie, about 70 people each).
FARN managers, and headquarter people work at full time for the FARN. Other people (about 260) work 20 weeks a year for the FARN, and the rest of their time for the NPP in their usual job.

Q.No	Country	Article	Ref. in National Report
8	India	General	Section B.3.3.2 (Summary) Page 21

Question/ Comment: Development of website that gives all environmental radioactivity measurement taken by all licensees, institutional bodies and approved associations is appreciated.

Answer: France really appreciates this comment.

Q.No	Country	Article	Ref. in National Report
9	India	General	Section B.3.3.5 (Summary) Page 23

Question/ Comment: While recognizing the concept of 'hardened safety core' and also that hardened safety core is based on diversified structures and components, would appreciate if following clarifications can be provided.
i) What is the position of France with respect to accident/external event affecting multi-units at a site, as on Page 36, it is mentioned that '…….could affect all the facilities on a given site', whereas on Page 37, it is '………large-scale event affecting several facilities simultaneously'.

Answer: The operating experience from the accident at Fukushima Daiichi NPP demonstrates that multiunit accidents that were not widely considered are possible.
It is expected that a multiunit accidental situation would be induced by a common cause, thus external hazards or induced effects of external hazards shall be verified (heavy load drop, pipe rupture) as well as indirect outside effects (upstream or downstream dam rupture…). The impact of such conditions on hazardous activities that are nearby should also be checked.

Q.No	Country	Article	Ref. in National Report
10	Ireland	General	General

Question/ Comment: Ireland thanks France for its comprehensive national report which includes the perspectives of both the regulators and the operators.

Answer: France really appreciates this comment.

Q.No	Country	Article	Ref. in National Report
11	Russian Federation	General	General

Question/ Comment: The National Report doesn't describe the system of safe operation indicators for France's reactors; it is only mentioned. It is known that EDF has a well-established system, which covers nearly all areas of NPP operation.
Are there indicators that assess leak-tightness of the reactor coolant circuit; if yes, what are they (their calculation methodology)? Does

	EDF use the "chemical index", used by WANO, as a key indicator, which assesses water chemistry of the plant, or there is a different methodology (please, describe)?
Answer	Indicators relative to status of safety functions and barriers are used by EDF. Regarding leak-tightness of the reactor circuit, indicator used is primary leakage rate, which is evaluated by comparing entering and outgoing flows in primary circuit. EDF has used the WANO chemistry indicator (CPI3) until the end of 2013. EDF intends to implement an adapted version of the INPO chemistry indicator (CEI) in 2014.

Q.No	Country	Article	Ref. in National Report
12	Russian Federation	General	General

Question/ Comment	According to the IAEA's Power Reactor Information System (PRIS), in the France there are reactors with different fuel cycles (12, 14, 16, 18, 24) in operation. Is there experience in analyzing the indicators used for assessment of safe operation that is based on duration of the fuel cycle? If yes, please describe.
Answer	In France, EDF PWRs are operated with different core managements and with different cycles lengths (12, 16 and 18 months). Nevertheless, the safety rules and the safety methodologies used are the same for all the plants. The safety level in operation is globally the same for the different plants, independently of the duration of the fuel cycle. In fact EDF don't use any indicator for assessment of safe operation, based on the fuel cycle duration.

Q.No	Country	Article	Ref. in National Report
13	Russian Federation	General	Section 2.2

Question/ Comment	What does account for France's policy to decrease dependence on nuclear power from 75 % to 50 %?
Answer	The reduction of the reliance on nuclear energy from 75% to 50% by 2025 is an objective set as part of the energy policy in France and independently from ASN. With 50% of nuclear energy in its energy mix, France would still aim at achieving and maintaining a high level of nuclear safety for its nuclear installations.

Q.No	Country	Article	Ref. in National Report
14	Russian Federation	General	Appendix 5

Question/ Comment	Appendix 5 contains information on OSART missions conducted at French power units. The text of Appendix says that reports on outcomes of conducted OSART missions are freely accessible at the website: http://www.asn.fr/index.php/English-version/Professional-events/OSART-Mission-conducted-in-France. Regretfully, this website doesn't contain reports on latest OSART missions to NPPs Cattenom and Chooz. Please, describe briefly the results of OSART Missions conducted at Cattenom (14 November 2011 - 1 December 2011) and Chooz

	(17 June 2013 - 4 July 2013); in particular, please provide good practices identified as a result of the missions.
Answer	ASN makes OSART reports public as soon as they are made available by the IAEA and derestricted. The report of the OSART mission conducted at Cattenom in 2011 is available on our website. The report of the mission conducted at Chooz in the summer 2013 will be available in March 2014. The main conclusions of the mission in Cattenom are mentioned on page 1 and 2 of the report (they are too long to be reported here and it is difficult to summarize them since they are presented as a list of bullet points related to different topics).

Q.No	Country	Article	Ref. in National Report
15	Spain	General	3.3.5.-24

Question/ Comment	A site emergency management center will be part of the hardened safety core. Could you elaborate more on the design specifications of this center, in particular on the seismic design?
Answer	Main characteristics and Safety Functions of the ECC (Emergency Crisis Centre) 19 buildings have to be erected : one for every nuclear site, Main characteristics of the building : o surface of about 3000 m² on 3 floors, o reinforced concrete building , o main entrance at the first floor; the first floor level depends on the flooding level considered for each nuclear site, o design based on shallow foundations unless the in-situ soil material properties require deep foundations., o design capacity up to 120 persons, o designed to withstand a long lasting radiological crisis, o autonomy of 72h for electricity, water and food, o control room for all plant units supervision, to estimate the physical states of the units and the impact of radiological release on the environment, o HVAC with toxic and radiological filtration Design specifications of ECC The ECC is part of the ""hardened safety core"" (HSC) SSCs. The design requirements for HSC are detailed in: o The components of the ""hardened safety core"" are considered as important to safety and assigned to the so called ""IPS-NC"" classification, which corresponds to the third level in the international safety classification system (IAEA Guide referenced DS367). o The hardened safety core have to be :

- composed of a limited number of Systems, Structures and Components (reliability),
- protected against extreme earthquake, flood and tornado, explosion, lightning, extreme climatic conditions, wind, snow, accidental rain, hail storm, wind generated missiles...
- protected against the effects that could be induced by these hazards,
- operable even if all other components are out of service (e.g. dedicated electrical source and I&C),
- operable without any material or human support from the outside during 24 hours following the event until FARN set-up (Nuclear Rapid Intervention Force),

o All the Hardened Safety Core SSCs have a specific Safe Shutdown Earthquake called SND. The SND is 1.5 times higher than the SSE of the other safety systems of the plant. Note that the SND is defined with the respect of the SSE based on the site seismology. The 1.5 factor is of the order of magnitude of the margins between the Maximum Historically Probable Earthquake (MHPE) and the SSE.

Q.No	Country	Article	Ref. in National Report
16	Switzerland	General	Summary, P, 21/22

Question/ Comment	A PSR was conducted at Fessenheim in 2010 before Fukushima which gave rise to recommendation to allow Fessenhem continued operation by EDF. In the footnotes on pages 22 reference (2) has an annex concerning these measures post-PSR. It defines a spectrum of the design base earthquake. It does not seem that this spectrum was changed after Fukushima. Indeed reference (4) in the footnotes of page 22, shows in its annex the same spectrum. Were any attempts made to reassess the seismic hasard? Was there no re-assessment of the hazard levels after Fukushima?
Answer	The periodic safety review (PSR) process has been implemented in France since the early 90’s for NPP. The PSR process allows an in-depth conformity check and the implementation of additional safety requirements. This is a 10 year process. ASN made clear in its position regarding the conclusions of Fessenheim PSR that it didn't consider the experience feedback from the accident at the Fukushima Daiichi NPP which was analysed in a specific process (stress tests). This specific process allows the operating experience feedback from this accident to be anticipated and not directly related to the PSR process. The beacon measures consist in the implementation of a hardened safety core that relies on new or existing SSC. These measures are designed for beyond design external hazards to ultimately fulfil lessential safety functions. This limited scope requires the implementation of a limited amount of SSC’s for which beyond design conditions only apply to them. Regarding earthquake the hardened safety core design value will be probabilistically defined with a return period higher than 20.000 years.

	The resulting hazard to take into account for Fessenheim hardened safety core will be higher than the hazard defined for the last PSR (but that was enforceable to all seismically classified SSC used in the safety demonstration).

Q.No	Country	Article	Ref. in National Report
17	Switzerland	General	Section B, 3.2. P, 20

Question/ Comment	ASN will request a new IRRS mission in 2014 (8 years since the lase!). WANO will take place every 4 years. "You wrote that after 2013, all French nuclear power reactors will have been subject to an OSART mission. How many years did it take? How often has the operator to request an OSART or SALTO mission?"
Answer	28 years were needed to review the all French nuclear fleet under the OSART mechanism. On average, since the 90's, France have received one OSART mission every year.

Q.No	Country	Article	Ref. in National Report
18	Switzerland	General	Section B, 3.3.5. P, 24

Question/ Comment	a new on-site emergency plan (PUI) baseline has been deployed on all EDF sites since 15 November 2012 taking into account multiple facilities on a given site. Could you please inform on the on-site emergency plan (PUI) for sites having several installations (like research reactors), each operated by another operator?
Answer	For the sites including several installations of the same operator like CEA or AREVA sites, there is only one site emergency plan (PUI), common for all installations of the site. For the sites with several installations of different operators, each operator has one PUI for its own installation and then the different operators have to prepare conventions defining coordination arrangements in case of an emergency.

Q.No	Country	Article	Ref. in National Report
19	Ukraine	General	para 6.3.1.3 page 38

Question/ Comment	To meet the requirement of ASN, based on the stress-tests' results EDF developed and submitted for the Regulator's review a feasibility study of the technical system that would prevent radioactive contamination of the ground water in case of severe accident with reactor vessel melt through. Has the Regulator (ASN) completed review of the Feasibility Study materials? In case it has, what was the result of the review? Can you give the brief description of this system?
Answer	ASN required EDF a feasibility study for the installation or renovation of a geotechnical containment or equivalent technical measure

	to prevent the transfer of radioactive contamination to groundwater and, by means of underground flow, to the surface waters, in the event of a severe accident leading to corium melt-through of the vessel. EDF provided studies that are under analysis by the ASN technical support organisation. ASN's TSO should provide its position to ASN by June 30th 2014.		
Q.No 20	Country Ukraine	Article General	Ref. in National Report To the whole Report
Question/ Comment	The concept of «hardened safety core» (Requirement ECS-01) is unique, as has been repeatedly emphasized at European level (during "stress-tests" and National Action Plans peer reviews). Is it possible to provide more detailed information about the implementation of «hardened safety core», except that all licensees in 2012 submitted to ASN the specification for «hardened safety core», and 13.12.2012 Advisory Committee for reactors has discussed this issue and provided their suggestions? According to the French National Action Plan based on stress-tests results, all measures related to the «hardened safety core» implementation, except ECS 18 (Additional electrical power supply means), shall be implemented (in full scope) by the end of 2013.		
Answer	The hardened safety core relies on the implantation of additional SSC's or existing SSC's which are designed or checked against beyond design conditions (external hazards and a plant situation after this external hazard, with consideration of induced effects). The global function of the hardened safety core is to guarantee ultimately basic safety function with reinforced means (criticality control, residual power evacuation, radiological confinement). This hardened safety core relies on additional means. For reactors, this additional means comprises mainly: • Bunkered diesel generator • New ultimate heat sink • New steam generator water feeding system • Reinforced I&C for the steam generator and steam released valves • Additional primary water feeding circuit • Containment sump heat exchanger and related out-containment cooling system. • Related I&C • Reinforced primary pump seal protection system • Containment isolation system… These SSC need the operation of existing systems such as hydrogen recombiners that are in place on French plants for years. For spent fuel pools, mainly: • Bunkered diesel generator (same as reactors)		

• New ultimate heat sink (same as reactors)
• Related I&C
• Reinforced Water feeding circuits
In addition, as part of the hardened safety core, an additional on site emergency response centre will be implemented to cope with multi units accidental situations.
The implementation of the hardened safety core requires that existing SSC that have safety functions under specific conditions are checked regarding these conditions (reactor containment, PARs…), spent fuel pool structural integrity (under extreme hazard and induced effects such as heavy load drop).
This hardened safety core that relies on fixed means is also designed to be compatible and to house plugging systems to be supported if necessary by mobile means provided by some national repository.
On January 2014 ASN issued new resolutions to EDF related to the design and the implementation of the hardened safety core. Once translated in English, these decisions will be available on ASN web site: http://www.asn.fr.
The implementation of the most significant measures related to the hardened safety core (typically Bunkered diesel generator, New ultimate heat sink, additional on site emergency response centre) is forecasted by 2020 for the latest sites.

Q.No	Country	Article	Ref. in National Report
21	Belgium	Article 6	chap. C, sec. 6.3.1.3, pages 36-37

Question/ Comment -
Among the resolutions issued by ASN on 26th June 2012, the creation of a "hardened safety core” of material and organisational measures is requested for managing the basic safety functions in extreme situations for all NPPs (ECS-01). The report provides general requirements on what the "hardened safety core” has to withstand and resist to, and on diversified heat sink and electrical sources it has to be equipped with. What are the “basic safety functions” that the "hardened safety core” is required to manage (for example, water makeup to the primary circuit, to the secondary circuit, to the spent fuel pools)?

Answer The hardened safety core relies on the implantation of additional SSC’s or existing SSC’s which are designed or checked against beyond design conditions (external hazards and a plant situation after this external hazard, with consideration of induced effects).
The global function of the hardened safety core is to guarantee ultimately basic safety function with reinforced means (criticality control, residual power evacuation, radiological confinement).
This hardened safety core relies on additional means. For reactors, this additional means comprises mainly:
• Bunkered diesel generator
• New ultimate heat sink
• New steam generator water feeding system
• Reinforced I&C for the steam generator and steam released valves

• Additional primary water feeding circuit
• Containment sump heat exchanger and related out-containment cooling system.
• Related I&C
• Reinforced primary pump seal protection system
• Containment isolation system…

These SSC need the operation of existing systems such as hydrogen recombiners that are in place on French plants for years.

For spent fuel pools, mainly:

• Bunkered diesel generator (same as reactors)
• New ultimate heat sink (same as reactors)
• Related I&C
• Reinforced Water feeding circuits

In addition, as part of the hardened safety core, an additional on site emergency response centre will be implemented to cope with multi units accidental situations.

The implementation of the hardened safety core requires that existing SSC that have safety functions under specific conditions are checked regarding these conditions (reactor containment, PARs…), spent fuel pool structural integrity (under extreme hazard and induced effects such as heavy load drop).

This hardened safety core that relies on fixed means is also designed to be compatible and to house plugging systems to be supported if necessary by mobile means provided by some national repository.

On January 2014 ASN issued new resolutions to EDF related to the design and the implementation of the hardened safety core. Once translated in English, these decisions will be available on ASN web site: http://www.asn.fr.

The implementation of the most significant measures related to the hardened safety core (typically Bunkered diesel generator, New ultimate heat sink, additional on site emergency response centre) is forecasted by 2020 for the latest sites.

Q.No	Country	Article	Ref. in National Report
22	Belgium	Article 6	chap. C, 6.1.2 page 28-29

Question/ Comment	- Ii is claimed that the research reactors are subject to the same regulations as nuclear power reactor ([…] the safety of French research reactors, which are subject to the same regulations as nuclear-power reactors). But a little later it is stated that the analysis of their safety case and the steps taken to guarantee it are the result of a “graduated approach”. Does it mean that (1) research reactors are subject to the same constraints as nuclear power reactors (regulations, IAEA guides for power reactors, WENRA reference levels, etc.) but that they must justify, items per items, the degree to which they will fulfil these constraints or (2) that some rules (regulations, guides, references, etc.) are considered ,a priori, as non-applicable ?

Answer	Regarding regulatory framework, the same constraints apply to all nuclear facilities. The WENRA reference levels declined in the French regulation (Order of 7 February 2012 setting the general rules relative to base nuclear installations, and the resolutions that specify or will specify this order) are therefore applied to all nuclear facilities. In Article 1.1 of the 7 February 2012 Order, the “graduated approach” is defined as : “Their application [of the general rules relative to base nuclear installation] is based on an approach that is proportional to the extent of the risks or drawbacks inherent to the installation. It takes into consideration all the technical aspects and relevant organisational and human factors.” In its Safety Analysis Report (SAR), the licensee demonstrates that it reaches the level of safety and the requirements of the regulation. The evaluation performed on the SAR by the regulatory body and its TSO also illustrated this graduated approach. For instance, for the periodic safety review that is required for each nuclear facility every ten year, the evaluation can be performed only by our TSO for installations with less risks, or external permanent group of experts gives its opinion for installations with more risks. Another example is the post-Fukushima complementary safety assessments (CSA). In this process, all the French facilities were divided into three categories of decreasing priority, depending on two main factors: on the one hand, their vulnerability to the various phenomena that led to the Fukushima Daiichi NPP accident, and on the other hand, the amount of radioactive elements that would be dispersed in the event of a failure of the safety functions. On the 79 high-priority facilities, only five of them are research or experimental reactors (including two currently shutdown or in decommissioning) and their operators (the “Comissariat à l’Energie Atomique et aux Energies Alternatives” (CEA) and the “Institut Laue Langevin”) submitted their reports to the ASN on September 15th 2011. Concerning the lower-priority facilities, including three other facilities (two research reactors operated by the CEA and a facility operated by ITER Organization) the deadline was September 15th 2012. Finally, the remaining facilities were not asked to submit a report yet, but they will have to do it later, mainly on the occasion of their next periodic safety review.

Q.No	Country	Article	Ref. in National Report
23	Belgium	Article 6	chap. C, 6.3.2.1 page 40

Question/ Comment	- Jules Horowitz reactor: requirements for extreme event are they the same as those imposed on power reactors ? In particular, do you use the same return periods (design basis and beyond design basis) as those chosen for nuclear power plants ?
Answer	Requirements for extreme event are the same as those imposed on power reactors. Indeed, fundamental safety standards as RFS n°2001-01 regarding the earthquake risk which sets the return period for example, apply as well to research reactor as to power reactor for design basis and for the periodic safety review that is required for each nuclear facility every ten year. Furthermore, after the accident at Fukushima Daiichi NPP, complementary safety assessments have been performed on the same bases for all BNIs (nuclear

	power plant and RJH being studied in the same batch of facilities). In addition to the common request applicable to all BNIs, it led to the definition and implementation of a "hardened safety core" of material and organisational measures to control the fundamental safety functions in extreme situations.

Q.No	Country	Article	Ref. in National Report
24	Canada	Article 6	Page 42, Section 6.4.1.4 Line 4

Question/ Comment	The report states that "…In 2010, ASN in particular requested that safety reassessment studies and the associated radiological objectives be considered in the light of the safety objectives applicable to new reactors", please clarify whether these ASN requirements are also applicable to Long-term Operation (LTO) or, as well, to any licence renewal? If yes, how are they implemented in the licensing basis of operating NPPs?
Answer	In France in the licence of the existing NPPs there is no time limit for operation, therefore there is no licence renewal. However, these objectives (consisting in approaching as most as possible the objectives given to new reactors) are applicable for LTO, witch means in France every PSR beyond 40 years.

Q.No	Country	Article	Ref. in National Report
25	China	Article 6	section 6.4.1.1

Question/ Comment	Description In section 6.4.1.1£¬P42 : ¡°The guarantees obtained. ASN more specifically reissued its request for re-inspection every 5 years of the Tricastin 1 vessel, which comprises 20 defects under the liner and asked EDF to maintain or install heating of the safety injection system on the Tricastin 1, Fessenheim 2 and St Laurent B 1 reactors, in order to minimize vessel loadings in the event of an accident situation. Question: Please provide the information of the defects and the present progress to deal with it.
Answer	In 1999, defects were found in Tricastin 1 RPV. This RPV shows 20 undercladding cracks and their maximum dimensions are about 10mm high (through wall direction) and 50mm long. After discovering undercladding cracks in RPV noozles in the late 70's (1978), changes were implemented in manufacturing process (pre heating and post heating for example) and control were performed on some equipment in service (nozzles, Steam generators tube sheets…). After first analysis, RPV was supposed not to be affected by such defects. ASN have been requiring controls on some RPV since 1991 and the first undercladding cracks were found on RPV in 1993. Since 1999, which corresponds to the first 2nd ten yearly outage of French reactors. If a defect is found in a RPV in France, a specific justification should be performed. ASN performs its review, with its technical support IRSN, to ensure that all demonstration's step are conservative enough: Fluence estimation, RPV material embrittlement, Transient and loading definition, Mechanical calculations. The demonstration performed by the licensee has to be periodically reviewed: the last review date back to 1987, 1999, 2005 and 2010.In France, undercladding cracks are controlled every ten years (every 5 years at Tricastin 1). No undercladding crack, neither in a RPV or in a nozzle has ever evolved during operation. Undercladding cracks are manufacturing

defects but there are important topics for ageing since RPV's mechanical properties (and especially metal's toughness) are going lower while ageing.
RPV fitness for service demonstration is identified as a limitation for reactor service since French requirements are very strong and conservative (the vessel with highest RTNDT evaluated with FIS formula is about 80°C after 40 years operating and is at current demonstration's limit.

Q.No	Country	Article	Ref. in National Report
26	China	Article 6	section 6.3.1.4

Question/ Comment: Why a seismic interaction procedure is needed to protect non-earthquake seismic equipment£¿Is it an ASN requirement?

Answer: The "seismic interaction" procedure aims to prevent, in the event of an earthquake, seismic classified equipments (= targets) from being damaged by non-seismic classified equipments or structures (= aggressors).
These indirect (secondary) effects of seismic events have been examined as of the second 10-year outage of the 900 MWe reactors in the framework of the periodic safety reviews. Then, the initial approach was supplemented by the study of the potential damage the buildings of the nuclear island by the turbine hall.

During the inspections, ASN noticed that the seismic interaction approach was not correctly applied: for example an overhead crane was not in its position of garage and the operational documents did not explicitly mention the procedure. So, following the accident at Fukushima Daiichi NPP, ASN required to the licensee to take the necessary steps to prevent equipments whose operational availability is required for the safety demonstration (= seismic classified) from being damaged by other equipment items (that are not seismic classified) in the event of an earthquake, no later than 31 December 2012.
Where protections are needed, dispositions could be:
- Move the target or the aggressor,
- Insure a structural holding of the aggressor,
- Implementation of a protection of the target,
- Justification of the holding of the target under the effects of the aggressor by analysis or essays,
- Operational requirements on the equipments (ex: put the overhead crane in its position of garage).

Q.No	Country	Article	Ref. in National Report
27	Germany	Article 6	page 31- 32

Question/ Comment: The main changes made following adoption of the 900 MWe VD3 safety baseline requirements include:
[…]
- improved severe accident management, notably by increasing the reliability of the reactor coolant system depressurisation device

	with the pressuriser valves, even in the event of severe accidents generated by a station black-out situation.[…] What measures were taken to increase the reliability of the pressurizer valves for reactor coolant system depressurization?
Answer	The single action electromagnets to order the pressurizer valves are replaced by single action latching solenoids qualified by tests performed under severe accident conditions. In the event of the loss of power supply or damage of the electric cables, permanent magnets continue to maintain the valves in a open position to ensure the primary circuit depressurization. A new mobile equipment allows the activation of the electromagnets under severe accident conditions generated by a station black-out.

Q.No	Country	Article	Ref. in National Report
28	Germany	Article 6	6.3.1.1.1., 32

Question/ Comment	In a resolution of 4th July 2011, ASN issued forty additional requirements for Fessenheim reactor 1 following its third ten-yearly outage. These include the following two main requirements: reinforce the reactor basemat before 30th June 2013, to increase its corium resistance in the event of a severe accident with vessel melt-through. After review of the file submitted by EDF, ASN on 18th December 2012 authorised EDF to carry out the proposed modification subject to a number of additional conditions designed to ensure the radiation protection of the personnel carrying out the work and to ensure compliance with the safety objectives set. The modification is designed to increase both the thickness and the surface area of the corium spreading area in the event of a severe accident with vessel melt-through. The basemat was reinforced during the reactor outage in the spring of 2013 and no problems were encountered.[…] Please give a short description of concept and implementation of this measure. Were any analyses performed to assess the effectiveness of the reinforcement of the basemat?
Answer	The modifications consist firstly in thickening the basemat of the reactor cavity beneath the reactor vessel. The height available beneath the vessel being limited, a secondary spreading area has been defined in a room neighbouring the reactor cavity to further increase the basemat resistance. The final total surface for corium spreading is now 3 times larger than the initial zone beneath the reactor vessel. The second spreading area is thickened as the reactor pit. This spreading room and the reactor cavity are connected by a penetration which was drilled in the wall of the reactor cavity. These modifications have been done in addition to those which are achieved in the frame of the third ten-yearly safety reassessment to reduce the risk of severe accident with core melt. Two kinds of analyses have been performed to assess the effectiveness of the reinforcement of the basemat. First of all core concrete interaction calculations have been done to check that the resistance of the basemat is increased by both the increase of the thickness of the basemat and the corium spreading area. The calculations have been performed with the MCCI (Molten Core Concrete Interaction) code TOLBIAC-ICB [1] in different scenarios (e.g. with or without water). These calculations show that the modifications ensure that should a basemat melt-through occur, then it would be at least three days after the beginning of the accident. This ensures compliance with the safety objectives set. It was also demonstrated with THEMA [2] calculations that the corium fully spreads on the spreading

area that is connected to the reactor cavity.

[1] « Simulation of MCCI with the TOLBIAC-ICB code based on the phase segregation model » B. Spindler, B. Tourniaire, J.M. Seiler, Nuclear Engineering and Design, Vol. 236, 2006

[2] « The simulation of melt spreading with THEMA code : Part 1 : model, assessment strategy and assessment against analytical and numerical solutions », B. Spindler, J.M. Veteau, Vol. 236, 2006.

Q.No	Country	Article	Ref. in National Report
29	Germany	Article 6	6.3.1.3. p.37

Question/ Comment

[…] In 2012, the licensees sent ASN the content and specifications of a "hardened safety core" for each facility. On 13th December 2012, the Advisory Committee for reactors met to give its opinion on:

Requirement ECS-01:

[…]

- the objectives associated with the hardened safety core and its functional perimeter,
- the types and levels of initiating events considered when defining the hardened safety core,
- the choices adopted when considering the events that these initiating events induce on the facility and the hardened safety core,
- the implementation conditions for the hardened safety core, more specifically the facility states in which it can be used,
- the requirements associated with the equipment of the hardened safety core,
- the methods and criteria used to demonstrate compliance with the requirements,
- the integration of organisational and human factors for the implementation of the hardened safety core provisions,
- the emergency management provisions planned to meet the requirements of the hardened safety core.

ASN will issue an opinion on these points in 2013.

Please give an overview of the ASN opinion on the content and specifications on the hardened safety core.

Answer

On December 2012, the Advisory Committee gives its opinions on EDF hardened safety core and advised ASN for complements. During 2013 EDF provided complements that were addressed by ASN and its technical support organisation.

On January 2014 ASN issued new resolutions to EDF related to the design and the implementation of the hardened safety core. Once translated in English, these decisions will be available on ASN web site: http://www.asn.fr.

ASN resolutions supplemented EDF proposal on the hardened safety core, in particular regarding safety objectives, and they provide some complements regarding design specifications or functionalities.

Q.No	Country	Article	Ref. in National Report
30	Japan	Article 6	6.3.1.3, p36, p37

Question/ Comment	French report refers several ECS. Are ECSs regulatory requirements? How are ECSs defined in the French legislation, law, act, decree, order or others?
Answer	ECS were requested by ASN resolutions (legally binding documents) to the operators.

Q.No	Country	Article	Ref. in National Report
31	Japan	Article 6	6.3.1.3, p37

Question/ Comment	French report says, □gthe licensees sent ASN the content and specifications of a hardened safety core.□h What safety classes, qualification category or ESPN level do you expect to be assigned to the components which compose the hardened safety core? Safety classification and qualification class are referred in page 154 and 155.
Answer	The components of the ""hardened safety core"" are considered as important to safety and assigned to the so called ""IPS-NC"" classification, which corresponds to the third level in the international safety classification system (IAEA Guide referenced DS367). The SSC of the ""hardened safety core"" are subjected to the following requirements : - aptitude to carry out the needed mission in extreme accidental conditions - design and manufacture under quality assurance for the new SSC, - aptitude for periodic tests, - follow-up throughout all the installation lifetime.

Q.No	Country	Article	Ref. in National Report
32	Japan	Article 6	6.3.1.3, p37

Question/ Comment	French report says, □gThe equipment which is to be part of this hardened safety core must be designed to withstand major events on a scale far in excess of that considered.□h How far is enough in excess of that considered? Please show basic concept and technical bases of the determination.
Answer	To illustrate the meaning of “the equipment which is to be part of this hardened safety core must be designed to withstand major events on a scale far in excess of that considered”: - regarding the seismic issue, the hardened safety core will have to be designed to a hazard that is probabilistically defined with a return period higher than 20.000 years. The hardened safety core is thus designed to a higher seismic level than the original or revised design level for the rest of the installation. - the flood level is defined with significant margin compared to the flood derived from ASN guide on the flooding of nuclear installations published in 2013. The corresponding return period is < 10-4. - ASN asked the operator to define other beyond design external hazards for the hardened safety core (high wind, tornadoes, extreme temperature…). These values will be submitted by operators to ASN.

	In addition new SSCs shall be designed according to state of the art design code and existing one shall be verified according to methods that are usually accepted during period safety review assessments in France. These codes and methods include significant conservatisms.		
Q.No 33	Country Luxembourg	Article Article 6	Ref. in National Report page 32
Question/ Comment	Fessenheim 1 had works done on the basemat to increase its thickness and surface (as mentioned on page 32). As increasing the thickness of the basemat sounds rather complicated, could you elaborate how these works were realized?		
Answer	The thickness increasing of the basemat beneath the reactor vessel has been a challenging operation due to the high level of radioactivity, and presence of critical equipments in the working areas (In-core Instrumentation tubes and penetrations in the reactor vessel bottom head). The modifications consist firstly in thickening the basemat of the reactor cavity beneath the reactor vessel. The height available beneath the vessel being limited, a secondary spreading area has been defined in a room neighbouring the reactor cavity to further increase the basemat resistance. The final total surface for corium spreading is now 3 times larger than the initial zone beneath the reactor vessel. The second spreading area is thickened as the reactor pit. This spreading room and the reactor cavity are connected by a penetration which was drilled in the wall of the reactor cavity. These modifications have been done in addition to those which are achieved in the frame of the third ten-yearly safety reassessment to reduce the risk of severe accident with core melt. The implementation was based on largely automated tasks and operations performed by remotely controlled robots. All activities were supervised by remote video monitoring. The concrete was formulated to be self compacting. One of the key measure to secure the works was a sound preparation : o for example by performing the implementation several times on real size mock-ups of the working areas; o by working out alternative measures in case of unexpected issues, for example, duplication of the concrete batching plant in case of breakdown.		
Q.No 34	Country Russian Federation	Article Article 6	Ref. in National Report Subsection 6.3.1.2.2
Question/ Comment	This Subsection speaks of the steam generator replacement program (RGV) due to susceptibility of the steam generator tube bundles material to a number of corrosion phenomena, which lead to tube degrading. Is adjustment of water chemistry carried out at power units because of this problem? If yes, is it successful? What are types of water chemistry in use?		
Answer	Replacement of SGs equipped with tube bundle made of 600MA and 600TT are planned between 3rd and 4th ten-yearly outage due to		

Primary Water Stress Corrosion Cracking (PWSCC) in rolled transition region. Extensive preventive plugging has to be performed in order to prevent leakage during hydraulic test that has to be performed each ten years at a pressure of 207 bar. Adjustment of water chemistry is not linked to this tube degradation.

Q.No	Country	Article	Ref. in National Report
35	Slovakia	Article 6	p. 27

Question/ Comment: Could France indicate the increase of costs for safety upgrading measures after Fukushima?

Can you define impact of adopted safety measures after Fukushima accident on employment (number of employees) in French nuclear industry?

Answer: As indicated in 11.1.2 of the French report, EDF intends to invest a total of about 10 billion Euros in order to take into account the lessons learned from the Fukushima accident and to meet the ASN requirements.
The question about the impact on employment in the French Nuclear Industry cannot be answered within the scope of the convention.

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Q.No	Country	Article	Ref. in National Report
36	Slovenia	Article 6	6.4.1.4/42

Question/ Comment: Do you have a final high-level radioactive waste disposal site (e.g spent fuel)?
This is a precondition (IAEA) for continues reactor operations beyond 40 years.

Answer: The “Waste act” of 31/12/1991 defined three axis of research and studies in order to deal with management of high-level and intermediate level long-lived waste : separation and transmutation of long-lived radionuclide, long-term storage and deep geological disposal. After 15 years of studies, ASN issued a stand on the work that had been carried out. Moreover, a public debate has been held on this topic. Following these steps, the “Program act” of 28/06/2006 defined the deep geological disposal as the reference solution for management of these radioactive waste and required Andra to submit an application file for a deep geological repository in 2015. If

	authorization is granted, this repository should be commissioned in 2025. Thus, if there is no final high-level radioactive waste disposal site in France, the process that is necessary for the development of such an installation has been taken in place for decades.		
Q.No 37	Country Switzerland	Article Article 6	Ref. in National Report Summary
Question/ Comment	The report contains a list of post-Fukushima Daiichi actions. Could all safety improvement actions and activities planned up to 2013, be finished? Are the deadlines for scheduled actions in 2014 still up to date? If not please identify these actions and activities.		
Answer	Post-Fukushima Daiichi operating experience feedback actions that are due by operators in 2013 and in 2014 on a regulatory basis have been identified in ASN decisions issued on June 26th 2012. Responses by operators are currently conforming to requirements. Implementations of modifications are monitored by ASN from their definition to their implementation. ASN inspectors regularly proceed to verifications in their routine inspection process. No drift is observed in the operator's action plan. Though, it remains a collective challenge to achieve to implement all the safety improvements, in due time, with a rigorous monitoring and without adverse impact on other essential activities (operational safety).		
Q.No 38	Country Switzerland	Article Article 6	Ref. in National Report P, 27
Question/ Comment	What will be the overall safety benefit for the French NPPs (e.g. core damage frequency, large release frequency) after implementation of all requested safety improvements?		
Answer	Even if all French NPP belong to standardized plants series, their overall safety level depends on sites specificities, especially regarding external events. Consecutively, the benefits of post-Fukushima modifications (which aim to cover beyond design conditions including extreme external hazards) depend on these specificities. A this stage, first simplified assessments (extended to seismic and external flooding risks) performed by EDF indicate that post-Fukushima modifications allows us to go towards ambitious probabilistic targets like a CDF lower than 10–5/r.y (INSAG-12 targets for new NPP).		
Q.No 39	Country United Arab Emirates	Article Article 6	Ref. in National Report 32
Question/ Comment	The National Report of France states that in a resolution of 4th July 2011, ASN issued forty additional requirements for the Fessenheim reactor 1 following its third ten-yearly outage VD3, including: a) reinforcement of the reactor basemat to increase its corium resistance in the event of a severe accident with vessel melt-through and b) implementation of emergency technical measures for long-term removal of residual heat in the event of loss of the heat sink. Could France give details on the likely effectiveness of these measures to prevent basemat melt through and to prevent the release of non-condensable gases from the containment to the		

	atmosphere?
Answer	a) The modifications consist in a thickening of the basemat of the reactor cavity and the foundation of a spreading area in a room neighbouring the reactor cavity. This spreading room and the reactor cavity are connected by a hole which was drilled in the wall of the reactor cavity. These modifications have been done in addition to those which are achieved in the frame of the third ten-yearly safety reassessment to reduce the risk of severe accident with core melt. Two kinds of analyses have been performed to assess the effectiveness of the reinforcement of the basemat. Concerning the release of non-condensable gases, studies have shown that the mass flow rate of non-condensable gases is related to the ablation rate of concrete and thus is mainly controlled by the residual power (the increase of the surface of interaction decreases the heat flux towards the concrete and thus the ablation rate). Thus the modifications which have been done do not lead to an increase of the release of non-condensable gases during the molten core concrete interaction. b) Regarding the implementation of emergency measures for long term removal of the residual heat in the event of loss of the heat sink, the implementation of new source of water (mainly pumping in aquifers, or new storage tanks) leads to a diversification of the heat sink which ensures the long term feeding of the steam generators with a new diversified EFWS and the removal of the residual power in this situation.

Q.No	Country	Article	Ref. in National Report
40	United Arab Emirates	Article 6	37

Question/ Comment	The National Report of France describes a new requirement for "hardened safety core" of material and organisational measures for managing the basic safety functions in extreme situations.. The report states that in 2012, the licensees sent ASN the content and specifications of a "hardened safety core" for each facility and that in December 2012, the Advisory Committee for reactors met to give its opinion The National Report states that the ASN will issue an opinion on these points in 2013. Could France elaborate on ASN's opinion on this matter?
Answer	On December 2012, the Advisory Committee gives its opinions on EDF hardened safety core and advised ASN for complements. During 2013 EDF provided complements that were addressed by ASN and its technical support organisation. On January 2014 ASN issued new resolutions to EDF related to the design and the implementation of the hardened safety core. Once translated in English, these decisions will be available on ASN web site: http://www.asn.fr. ASN resolutions supplemented EDF proposal on the hardened safety core, in particular regarding safety objectives, and they provide some complements regarding design specifications or functionalities.

Q.No	Country	Article	Ref. in National Report

41	United Arab Emirates	Article 6	43

Question/ Comment	The National Report mentions that EDF has stated that it wishes to extend the operating life of its reactors beyond 40 years and that ASN will in 2013 rule on the orientations of a study programme dedicated to the reactor operation extension. Could France elaborate on this matter and give details on the intended the criteria that ASN will apply to decide on the adequacy of the licensees proposals?
Answer	ASN considers that the program of works and studies proposed by EDF to support LTO is acceptable, but needs to be enhanced on some issues that ASN pointed out in its letter (this letter is currently being translated into English and will be uploaded on ASN's website once it is done). The reassessment of the safety level should lead to reduce as much as possible the radiological impact of nuclear accident considered in the design of the plant, introduce strong measures to prevent and limit the consequences of severe accidents, and improve the level of safety of spent fuel pools. ASN with the help of its technical support (IRSN) will assess if EDF's propositions are sufficient to reach as much as reasonably possible safety objectives given to new reactors. The criteria to decide on the adequacy of the licensee proposals are not precisely defined yet. Pursuant the November 2010 WENRA statement, ASN aims at an ambitious safety level taking the safety objectives defined for the new reactors as a reference.

Q.No	Country	Article	Ref. in National Report
42	United Kingdom	Article 6	Page 28, Section 6.1.1.2

Question/ Comment	Please clarify whether filtered containment venting is to be fitted to Flamanville 3 and to the other classes of French NPP
Answer	On 26th June 2012, ASN issued resolutions setting some complementary requirements (resolution ECS-28) : « Before 30 June 2012, the licensee shall present ASN with the systems specified in the preliminary safety analysis report, or any systems to be added and constituting a part of the hardened safety core in order to ensure control of pressure in the containment in the event of a severe accident. Within the same time-frame, the licensee shall send ASN a study of the advantages and drawbacks of the various possible systems. » EDF answered this resolution by presenting 3 potential solution in order to control the pressure in the containment. EDF chose to valorise the use of a mobile devices (pump and electric supply) in addition to CHRS system in order to control the pressure in the containment system. EDF rejected the possibility to use a filtered containment venting system, because it was thus impossible to reduce sufficiently the radiological releases to match the radiological targets set for FA3. In the GPR meeting of December 2012 on the hardened safety core, the IRSN didn’t object the solution proposed by EDF. ASN

doesn’t have officially taken position on the study transmitted by EDF, but considers that preventing any radiological release must be the priority. ASN considers the CHRS system sufficiently robust to control the pressure in the containment system :
• It is part of the hardened safety core,
• the water supply is sufficiently diversified (ultimate heat sink part of the hardened safety core, which water comes from the top of Flamanville cliff)

For the other classes of French NPP, ASN asked EDF to study the possibility to remove the heat from the containment without the use of filtered containment venting system. In the case that wouldn't be feasible, ASN would request a reinforcement of the existing equipment.

Q.No	Country	Article	Ref. in National Report
43	United Kingdom	Article 6	Page 43, Sec 6.1.4 and p25 Sec 3.4.1

Question/ Comment: The report states that ASN will rule in 2013 on a study and work programme proposed by EDF to support extending the operating life of reactors beyond 40 years. Please outline what ASN’s position is on the proposed programme.

Answer: ASN considers that the program of works and studies proposed by EDF to support LTO is acceptable but needs to be enhanced on some issues that ASN pointed out : identifying aging phenomena must be completed; a robust justification of vessels' strength beyond 40 years must be provided; proposals for improving the security level of spent fuel pools must be made; the radiological impact of accidents must be reduced as much as possible. The letter that reflects ASN's detailed position is currently being translated into English and will be uploaded on ASN's website once it is done.

Q.No	Country	Article	Ref. in National Report
44	United Kingdom	Article 6	Page 37, Section 6.3.1.3

Question/ Comment: ASN were scheduled to give an opinion on the hardened safety core proposals in 2013. Please provide an update. Also does the proposed emergency management centre have a remote monitoring and control capability?

Answer: On December 2012, the Advisory Committee gives its opinions on EDF hardened safety core and advised ASN for complements.
During 2013 EDF provided complements that were addressed by ASN and its technical support organisation.
On January 2014 ASN issued new resolutions to EDF related to the design and the implementation of the hardened safety core. Once translated in English, these decisions will be available on ASN web site: http://www.asn.fr.
ASN resolutions supplemented EDF proposal on the hardened safety core, in particular regarding safety objectives, and they provide some complements regarding design specifications or functionalities.

In addition, the emergency management centre will have some remote monitoring capabilities but no remote control capabilities.

Q.No	Country	Article	Ref. in National Report

45	Canada	Article 7.1	Page 175, Section 7.3.1.2, Paragraph 2

Question/ Comment: Can ASN elaborate on which basis it determines what “specific cases” are to be examined in order to ensure compliance with Radiation Protection Regulations and how does it determines the criteria common to all operators for the notification of radiation protection events?

Answer: The regulation of radiation protection is the same for all the operators ; the limitation of the worker's radiation dose is the same and the radiation protection events are the same for all nuclear activities.

Q.No	Country	Article	Ref. in National Report
46	Slovenia	Article 7.1	7.3.3.3./58

Question/ Comment: We were not in position to read the accurate text of Article L.591-5 of the Environment Code and to learn more on definition of »significant event« but if the definition or description is reproduced in the first paragraph of Chapter 7.3.3.3. of the Report it seems that a number of immediate reports – 830 – is very high and it is higher every year.

Although its written down that the review and assessment of these events by ASN and IRSN are described in detail in Chapter 19.7 we would appreciate if you can comment the number of reporting events – having in mind that those are only events, where the licensee is required to notify immediately.

Answer: The number of significant events (ES) notified by EDF in 2012 is increasing from the previous years. These events include safety, radioprotection and environmental issues. In 2012, 95 events were rated level 1 on the INES scale, one level 2. 734 event were below INES scale (level 0).
This increase is mainly related to radioprotection (ESR) and to a lesser extent to safety (ESS). The increase of the ESR is approximately 20% since 2011. It is mostly related to industrial radiography operations, and due to no compliance with technical checks and the calibration of the equipment (radioprotection mobile equipment and zoning).
Regarding the ESS, the increase is approximately 10% since 2011 and mainly due to non-compliance with the instructions in the maintenance operations (poor quality in the risk analysis).
In 2013, The number of significant events (ES) notified by EDF is slightly decreasing compared to 2012. 82 events were rated level 1 on the INES scale, one level 2 (radiation protection related event). 734 event were below INES scale (level 0).

Q.No	Country	Article	Ref. in National Report
47	Spain	Article 7.1	46

Question/ Comment: Monitoring of outside contractors can no longer be subcontracted, but the licensee can obtain assistance.7.1.3.1.2.2. Could you give additional information on this issue?

Answer: This provision relates to Important Activity for Protection. Monitoring the execution of Important Activities for the Protection

	achieved by an outside operator must be exercised by the licensee, who can not be entrusted to a provider. However, in particular cases, the licensee may be assisted in this surveillance, provided you keep the skills to ensure mastery. The licences ensures that those organizations that have expected competence, independence and impartiality necessary to provide the services.		

Q.No	Country	Article	Ref. in National Report
48	Turkey	Article 7.1	46

Question/ Comment	It is stated in the report that according to the Order of 7th February 2012 the most fundamental technical capabilities must be held by the licensee or one of its subsidiaries. What are these most fundamental technical capabilities?
Answer	The licensee must be able to ensure monitoring of activities it implements (design, construction, operation, making the final stop). As such, the licensee must be technically capable of taking any decision regarding the operation of the facility and meet the demands of ASN.

Q.No	Country	Article	Ref. in National Report
49	Turkey	Article 7.1	47

Question/ Comment	It is stated in the report that Order of 7th February 2012 enables the application of some ICPE ministerial orders to the BNIs. Could you please explain whether BNIs were in ICPE lists? If so why there is a need to enable the application of ICPE ministerial orders through Order of 7th February 2012?
Answer	INB are not ICPE. There is a specific regulation BNI (Title V of Book 9 of the Environmental Code), however in some areas the rules are the same as for the ICPE. In this case, the rules imposed on ICPE are applied but for this, there must be a BNI text provided (order of February 7, 2012).

Q.No	Country	Article	Ref. in National Report
50	Ukraine	Article 7.1	page 44

Question/ Comment	French legislative basis on the nuclear installations (BNIs) safety is constantly being improved, including harmonization with WENRA RLs and taking into account lessons learned from the "Fukushima-Daiichi" accident (for instance, "Order of 7th February 2012" and ASN draft resolutions, updated in accordance to "Order of 7th February 2012," which are currently being discussed with stakeholders). Does the Regulatory body have an intention to revise existing regulation or to develop new requirements for the implementation of the new WENRA publication "Report. Safety of new NPP designs" (March 2013)? Are there any plans to incorporate the provisions of the WENRA document "Position paper on Periodic Safety Reviews (PSRs) taking into account the lessons learnt from the TEPCO Fukushima Daiichi NPP accident" (March 2013) in the draft ISN resolution «Periodic safety review», which is currently being discussed with stakeholders?
Answer	ASN is indeed developing complementary new regulations and guidance addressing periodic safety reviews and safety of NPP design.

The WENRA report of March 2013 is taken into consideration for writing the guide dealing with the safety of NPP design.
The resolution dealing with the PSR process will be consistent with the version 2008 of WENRA RLs (Issue P). It is also overall consistent with the referred WENRA document that should result in a new version of WENRA RLs Issue P (currently being published for comments).
Some of the lessons learnt from Fukushima Daiichi NPP accident, more precisely safety provisions to face extreme external hazards will be addressed by these two documents.

Q.No	Country	Article	Ref. in National Report
51	United Arab Emirates	Article 7.1	48

Question/ Comment: Resolutions, guidelines, and policy notes are three types of regulatory documents that can be issued by ASN. Resolutions are said to be intended to clarify decrees and orders approved by Minister in charge of nuclear safety and radiation protection. Can France elaborate on the practical meaning of issuing resolutions and the means envisaged to enforce the resolutions if breached by licensees?

Answer: In policy notes ASN provides guidance in its fields of competence: regulation, enforcement and sanctions, oversight, transparency, international relations, management of radiological emergency, dismantling and decommissioning BNI in France. They are elaborated to publicize and explain the doctrine of ASN and are not binding.
Additional ASN decisions supplement government regulation. The licensees must comply with these decisions. ASN may give notice to the licensees to respect and enforce for failure.
The guides are not mandatory and ASN can not sanction a licensee who does not follow. Guides are documents to assist operators recommendations. The licensee may propose another way to achieve safety goals.

Q.No	Country	Article	Ref. in National Report
52	United Arab Emirates	Article 7.1	49

Question/ Comment: It is understood that development of technical standards and codes is the responsibility of the industry. Does the licensee propose, or does the Authority specify, the codes and standards that are adopted for a given BNI? If the former, how does the licensee notify the Authority of its proposal and how does the Authority review and accept the licensees’ choices?

Answer: The safety authority does not specify the codes and standards used for a given BNI. The licensee mentions in the Safety Analysis Report (SAR) the codes he adopts for a given project of constructing or modifying a BNI. This SAR is part of the file that the licensee transmits to the safety authority with the application for a license to create or modify a BNI or, in cases of lesser importance, with the notification of a BNI modification, There is no specific procedure for accepting the technical codes and standards. The safety authority may submit the codes to a technical examination by its TSO or by its standing expert committees. In such a case, following the examination, the safety authority transmits its observations and requests to the licensee.

Q.No	Country	Article	Ref. in National Report
53	United Kingdom	Article 7.1	Page 57, Section 7.3.3.1

Question/ Comment	ASN Inspectors swear an oath. What do the Inspectors promise to do? What legal basis does the oath have? What measures would be taken if an Inspector failed to live up to the promises made in the Oath?
Answer	The nuclear safety inspectors who fail to live up to the promises made in the Oath could have an administrative and penal sanction. The administrative sanction consist of withdrawing the qualification as an inspector. The penal sanction (Articles 226-13 of penal code) consisting of the disclosure of secret information by an inspector is punished by one year's imprisonment and 15,000 euro of penalties.

Q.No	Country	Article	Ref. in National Report
54	United States of America	Article 7.1	7.1.3.3

Question/ Comment	The report states that ASN has developed basic safety rules on various technical subjects concerning both PWRs and other BNIs. The report also states that these basic safety rules are not “strictly speaking regulatory texts.” (1) Please describe whether (and the extent which) the public is involved in the development of these basic safety rules. (2) Also, please describe the steps a licensee must take to demonstrate that an alternative to the basic safety rule will attain the safety objectives that the original basic safety rule establishes.
Answer	1 - RFS are currently replaced by ASN guides. ASN when preparing a guide, shall submit the project to the public for comments. 2 - There is no set procedure. The operator shall submit a report setting out how it intends to implement the safety objectives identified in the guide. ASN examines the report with the support of its TSO (IRSN) and exchange with the operator. The operator must have the agreement of ASN in order to implement the means envisaged.

Q.No	Country	Article	Ref. in National Report
55	Japan	Article 7.2.1	p46

Question/ Comment	--Order of 7th February 2012-- French report says, □gThis order details a large number of requirements and provides a legal basis for several of the requirements expressed by ASN.□h Does this order incorporate lessons learned from Fukushima accident, such as, severe external hazards, hardened safety core and emergency response system?□@If no, please show your plan?
Answer	The order is applicable to all nuclear installations, i.e. from NPP and nuclear fuel cycle facilities to waste repositories and industrial irradiator. It therefore includes general expectations, to allow case by case implementation considering the facility (graded approach). The order incorporates some key requirements in relation to TEPCO Fukushima Daiichi NPP accident. More specifically, the need to consider external natural hazards such as earthquake and flooding (including its dynamic effects) and plausible combination of hazards are mentioned. The level of hazards to be considered is however not set in the order as these are site dependent information. Another

provision deals with the need to practically eliminate events which would results in large radioactive releases or early radioactive release. The need for an emergency response plan at each nuclear installation, as well as its regular review (3 years max), are also set. The order does not requires the implementation of a hardened safety core. It however requires that adequate confinement and core/spent fuel cooling are ensured at any time.

Q.No	Country	Article	Ref. in National Report
56	United Arab Emirates	Article 7.2.1	51

Question/ Comment: Please further explain the authorisation process for the operations phase with reference to the EPR at Flamanville. For instance, after obtaining a DAC, is the operator required to obtain a further authorisation to commission and operate the facility? If so, what information must the operator submit to the Authority to demonstrate that the facility has been constructed in accordance with the requirements and can be operated safely?

Answer: Yes, after the DAC, the licensee must obtain a commissioning permit. The license is issued by ASN on the basis of a folder defined by article 20 of BNI decree :
o the safety report,
o the general operating rules,
o a study of the facility's waste management,
o the On-Site emergency Plan (PUI), the decommissioning plan and
o update of the impact study of the plant year.

After checking that the installation complies with the objective and rules specified in the Environmental Code and its implementing texts, ASN authorises the commissioning of the plant and communicates this decision to the Minister responsible for nuclear safety and to the Prefect.

Q.No	Country	Article	Ref. in National Report
57	United Arab Emirates	Article 7.2.1	51

Question/ Comment: Please give some examples of the sort of "non-signficant" modifications that the licensee may make and examples of those which require regulatory approval?

Answer: In application of the essential principle fixed in article L 593-6 of Environment Code, the operator of a research reactor, as any of BNI, is responsible for the safety of the installation. Furthermore, in application of articles 26 or 31 of the "decree 2007-1557 of 2 November 2007 concerning basic nuclear installations and the supervision of the transport of radioactive materials with respect to nuclear safety", when the operator envisages a modification to the installation such as to affect the interests mentioned in article L. 593-1 of the environment code, it sends notification to the regulatory body, forwarding a file containing all useful justification data, in particular the necessary updates of the elements in the authorisation decree or installation commissioning files and, in the event of a

modification to the on-site emergency plan. The operator may not implement its project before a new authorisation if the modification is significant (like a rise in its maximum capacity or an operation with criticality risk for examples) or in other case, expiry of a six-month period, barring express approval by the regulatory body, which can also extend this period if it considers that further review or issue of additional requirements is necessary.
For minor modifications (like for an experimental device), the regulatory body may exonerate the operator from the notification procedure provided that the operator sets up a system of internal checks offering sufficient guarantees of quality, independence and transparency.

Q.No	Country	Article	Ref. in National Report
58	United Kingdom	Article 7.2.1	Page 45, Section 7.1.3.1.2

Question/ Comment: The BNI order came that into force on July 1st 2013 is described in general terms in the National Report. Does the order make a legal requirement that any specific plant modifications be carried out? For example does the Order require a hardened safety core or the provision of a rapid intervention force such as FARN?

Answer: All facilities are concerned by period safety reviews for existing installations (art.9.4 of the Order of 02/07/12).
The ministerial order enforces high level safety principles and does require neither a hardened safety core nor provision of a FARN.
This kind of safety provisions (design or organisational) have been enforced by specific ASN's resolutions.

Q.No	Country	Article	Ref. in National Report
59	China	Article 7.2.3	section 7.3.2.1

Question/ Comment: Description in section 7.3.2.1 ,¡°Once the nuclear installation has started operating, all safety-related modifications made by the operator are subject to ASN approval. In addition to these checks necessitated by changes in installations or their operating procedures, ASN requires the operators to conduct periodic safety reviews (see ¡ì14.2.1.3), to reinforce safety requirements according to changes in techniques and doctrine on the one hand and to experience feedback on the other.
Question:
Are safety-related modifications reviewed in the same scope and standards with the initial review?
Are ASN review standards the same as IAEA standards?

Answer: Safety-related modifications are reviewed in the same scope and safety standards with the last periodic safety review.
IAEA texts (Basic Safety Standards), which describe safety principles and practices constitute references, which are widely drawn on in the drafting of French regulation. Several legislative and regulatory provisions relative to French nuclear installations are derived from or take up international conventions and standards, notably those of the IAEA.
Moreover, regulatory body's review and assessment are performed in line with the set of WENRA reference levels of January 2008 that are also derived from IAEA safety standards. There are about 300 common "reference levels" covering main relevant NPPs safety requirements, as safety management, installation design and operation, verification of safety, and emergency situations. French nuclear

regulation takes these reference levels into account as far as possible.

Q.No	Country	Article	Ref. in National Report
60	Korea, Republic of	Article 7.2.3	51

Question/ Comment	In page 51 of Article 7, it is stated that "A Local Information Committee (see ¡× 8.2.4) can be created as soon as the BNI creation authorisation application is submitted. In any case, it must be in effect once the authorisation has been issued". Please explain the hierarchy among the consultation and information organisations at a different level including EU, national, and regional level that regulator and operator are to consider in the regulation/project procedure.
Answer	There is no hierarchy to the phases of consultation and information. The regulatory body and the operator must make all the consultations required by the treaties (for example Article 37 Euratom) and the French law (for example public inquiry to a DAC, the CLI consultation for a draft limitation on discharges of BNI ...).

Q.No	Country	Article	Ref. in National Report
61	Poland	Article 7.2.3	p. 57

Question/ Comment	Does ASN send absolutely no information about the inspection programme to the operator, not even a vague scope?
Answer	The operators have no information about ASN inspection programme. About 15 to 20 inspections are carried out on each NPP a year. Some inspections are not announced, others are announced by a letter sent to the operator 2 weeks before, containing information about the topic and the agenda of the inspection.

Q.No	Country	Article	Ref. in National Report
62	Romania	Article 7.2.3	Section 7.3, page 82

Question/ Comment	It is mentioned that "in order to consolidate the credibility and quality of its actions, ASN has defined a qualification system for its inspectors based on recognition of their technical skills". Please provide more information on this qualification system for the inspectors or a reference where detailed information is already available (e.g. training curricula, competence evaluation methods, etc.).
Answer	ASN duty to oversight nuclear safety of NPP relies on inspectors empowered to monitor nuclear activities in France. To ensure that the inspectors achieve and maintain a high level of technical skills, a training programme has been implemented in ASN. It consists of an “enabling” training programme, a requirement for a minimum of experience (i.e. a minimum number of tutored inspections), a length of service of at least 6 months to be qualified as an inspector, and some mandatory training sessions to enhance technical skills in specific area. The training programme is continuously improved taking into account the experience feedback from the inspectors and the ASN needs. The implementation of this training programme is ensured by key indicators and an annual interview with the managers gives the

opportunity to sum-up the needs of each inspector.

Q.No	Country	Article	Ref. in National Report
63	Romania	Article 7.2.3	Section 7.3.2.1.2

Question/ Comment In Section 7.3.2.1.2 - Scheduled NPP Outages, it is mentioned that the "approval of the outage programme is the responsibility of ASN". Could you please provide more details on the following:

- how much time in advance of the planned outage is the outage programme submitted by the licensees for review and approval of the ASN?
- what are the review guidelines and criteria used by ASN staff in order to determine that an outage programme is acceptable and can receive regulatory approval?
- if the outage programme is approved by the ASN, does this mean that also the changes to the outage programme have to be approved by the ASN?
- what are the activities that ASN inspectors typically observe during a planned outage?

Answer The licensee submits an outage program 4 months before a planned outage. This program is presented to ASN during a specific meeting, 3 months before the outage. One of the objective of the program assessment performed by ASN is to check that the licensee takes into account in the program activities coming from feedback in other French nuclear power plants (additional controls, replacement of defective components...). ASN also checks that the deadline for processing to the correction of any identified deviations is acceptable.

Just before the beginning of the outage, the licensee send to ASN and update of the program, to take into account ASN assessment and/or to add into the program other controls. Most of the time, there are no important changes in the outages program.

At the end of the outage, before achieving core criticality, licensee needs an ASN authorisation. ASN grants or refuses this authorization based on the results of the oversight performed during the outage (e.g. maintenance activities performed, correction of the deviations detected, safety demonstration of the core, etc.)

ASN is currently drafting new regulation on nuclear power plant outages, this requirements will be required in this regulation.

Concerning the inspections performed by ASN during the outage, they are mostly dedicated to the maintenance activities. Welding activities, quality management of the activities,

EDF monitoring of the contactors, radiation protection of the workers are topics often inspected during outages.

Q.No	Country	Article	Ref. in National Report
64	Turkey	Article 7.2.3	50

Question/ Comment It is stated in the report a consultation process is done through European Commission with the EU member countries, pursuant to Article 37 of the Treaty Instituting the European Atomic Energy Community. Since France is a party to "The Convention on Environmental Impact Assessment in a Transboundary Context,1991 (Espoo convention)" and "Convention on Access to Information,

	Public Participation in Decision-making and Access to Justice in Environmental Matters (Aarhus Convention),1998", Is there a consultation mechanism with the non-EU countries?
Answer	Pursuant to Article R. 122-10 of the Environmental Code, when a project is likely to have significant effects on environment of another State, a member of the European Union or party to the Convention of 25 February 1991 on the assessment of the environmental impact in a transboundary context signed Espoo, or when notified by the State likely to be affected by the project, the competent authority in charge of the decision for authorization, approval or implementation of the project itself, notifies without delay the opening of the public inquiry the order and sends a copy of the file investigation. In addition, Art. R. 122-10 specifies the procedure to be implemented in the context of this consultation. Moreover, Article L. 120-1-1 of the Environmental Code provides a procedure for public participation in the development decisions affecting the environment. Any interested person may submit his observations.

Q.No	Country	Article	Ref. in National Report
65	Ireland	Article 7.2.4	7.2.5, p.51

Question/ Comment	Under Consultation of other countries of the European Union, France notes that it is required to consult the European Commission, under Article 37 of the Euratom Treaty, before licensing any discharges. Does this represent the full extent of France consultation with other countries of the European Union? If it is, how is this considered consistent with France’s policy on transparency given that the submissions to the European Commission under this process are confidential to the Commission and the assigned group of experts?
Answer	France responds to the Article 37 of the Euratom Treaty in accordance with the procedure set. The opinion of the commission is published in the Official Journal of European Union. France has a policy of transparency, as a result draft decisions related to the collection and discharges are submitted to comments from the public.

Q.No	Country	Article	Ref. in National Report
66	Korea, Republic of	Article 7.2.4	58

Question/ Comment	It is observed from the National Report that the number of significant events notified by the BNI licensees in 2012 is significantly higher than the data of the two previous years. Does the ASN have any explanation for the increase ?
Answer	The number of significant events (ES) notified by EDF in 2012 is increasing from the previous years. These events include safety, radioprotection and environmental issues. In 2012, 95 events were rated level 1 on the INES scale, one level 2. 734 event were below INES scale (level 0). This increase is mainly related to radioprotection (ESR) and to a lesser extent to safety (ESS). The increase of the ESR is approximately 20% since 2011. It is mostly related to industrial radiography operations, and due to no compliance with technical checks and the calibration of the equipment (radioprotection mobile equipment and zoning). Regarding the ESS, the increase is approximately 10% since 2011 and mainly due to non-compliance with the instructions in the

maintenance operations (poor quality in the risk analysis).
In 2013, The number of significant events (ES) notified by EDF is slightly decreasing compared to 2012. 82 events were rated level 1 on the INES scale, one level 2 (radiation protection related event). 734 event were below INES scale (level 0).

Q.No	Country	Article	Ref. in National Report
67	Luxembourg	Article 7.2.4	59

Question/ Comment: Concerning enforcement ASN has 2 instruments. The one seems to be rather rather weak (formal notice, prescription), the other one â€œsuspension of operationâ€ seems to be rather strong. Does ASN believe that the present enforcement instruments are sufficient, or that an additional instrument would be needed? Could an instrument such as direct administrative penalties close this gap between to strong and too weak?

Answer: Present enforcement measures have proven their efficiency, but ASN would like to introduce new enforcement measures like administrative penalties (daily penalty payment). This proposal could be introduced soon.

Q.No	Country	Article	Ref. in National Report
68	Turkey	Article 7.2.4	51

Question/ Comment: It is understood from the report that there are two main authorizations for BNIs: "The BNI creation authorisation" which is issued by a decree signed by the Prime Minister and countersigned by the Minister responsible for nuclear safety and "The Authorisation to Commission a BNI" which is issued by ASN. Which of these documents is the equivalent of a license and which of these documents holder is accepted as the licensee?

Answer:

Authorisation to Commission and Authorisation to Service are issued to the same person, the operator.
However, the Authorisation to Commission is the equivalent of a license and the operator must also obtain authorization service, in order to introduce radioactive materials in the facility.

Q.No	Country	Article	Ref. in National Report
69	Turkey	Article 7.2.4	7.3.2-54

Question/ Comment: It is understood from the report that ASN has the authority to suspend BNIs operation in case of a breach endangering nuclear safety. In French system, is it possible to revoke the authorizations? Who has the authority for revocation?

Answer: Only the Minister can revoke a BNI authorization because the authorization is given by ministerial decree. ASN must be consulted by the Minister before taking any decree or order relating to nuclear safety.

Q.No	Country	Article	Ref. in National Report
70	Korea, Republic of	Article 8.1	63

Question/ Since the IRSN operates as an independent public establishment, has it conducted any cooperative activity with or received contracts

Comment funded by nuclear promotion organizations (e.g. EDF, CEA)? And if there are such activities, is there any mechanism or provision in the ASN-IRSN five-year agreement to exclude possible conflicts-of-interest arising from those activities?

It is understood that the IRSN runs and implements research programs with a view to consolidating national public expert capability around the most advanced scientific knowledge. Does the IRSN have an independent research group? If not, the IRSN staffs are technical experts? Or researchers? Sometimes they can be a regulator?

Between two organizations of ASN and IRSN, are there any undue influences for making a technical decision?

Answer IRSN is a French public body with industrial and commercial activities, under the joint supervision of the Ministries of the Environment, Health, Industry, Research and Defence.
IRSN has three main missions:
Research and services of public interest, including public transparency
Support and technical assistance to the public authorities for civil or defence-related activities
Contractual assessment, study and measurement services for public and private organizations, both French and foreign.
Its legal status is to be an independent and autonomous public entity, “to the exclusion of any liability as an operator of nuclear facilities” according to article 1 of Decree No. 2002-254 dated 22 February 2002.
Since its creation, IRSN has been conducting research activities by its own, or within the frame of national or international collaborations in association with other entities, including: TSO, academic partners, and industrial organizations such as EDF, AREVA or the CEA in France. When performed with other entities, research activities are based on a jointly funded and partnership based programs. According to the above mentioned Decree, from February 22nd, 2002, IRSN has defined and applies a professional code of ethics, under the survey of a Group for ethics and compliance.
Furthermore, the ASN-IRSN five years agreement includes provisions for the conflict of interest aspects in its article n°6: Article 6 - Performance of actions Responsibility.
The actions defined in this agreement are performed in compliance with the ethics and quality rules and charters in force and under the entire responsibility of IRSN, whether performed by its own staff
or by subcontractors.”
IRSN has a code of ethics applicable to the Institute’s activities and to its staff members.

- IRSN has not set up, within its organisation, an independent research group.
- IRSN employs around 1,700 people, including many specialists, engineers, researchers, physicians, and technicians, as well as experts in nuclear safety, radiation protection and control of sensitive nuclear materials. IRSN staffs include both technical experts and researchers, and none of the staff member acting within IRSN frame of mission work as a regulator

(some of its staff might be seconded for a few years within ASN, but in this case, they are under the full authority of the French ASN). For the safety of civilian nuclear installations, IRSN only intervenes in the regulatory framework upon request of the French Nuclear Safety Authority.
IRSN does not operate as a regulator. In this frame, its public mission is to assess the safety demonstration performed by the industry. ASN is the entitled regulatory body for all activities related to the civilian nuclear field in France, and therefore performs the inspections, take the decisions, and has the enforcement power. IRSN performs, on demand, the safety assessment and analysis on a permanent basis the technical feedback of experience.
All decisions to be taken and which can be binding for any of the actors in the nuclear field are initiated and taken by ASN.

Q.No	Country	Article	Ref. in National Report
71	Romania	Article 8.1	8.1.2.3 Quality management system

Question/ Comment: What are the performance indicators used for measuring the effectiveness of ASN’s actions?

Answer: Performance indicators are defined for each process of the management system. Some indicators measure the respect of deadlines fixed by regulations or by ASN. Some indicators count the number of products that come out of a process. Other indicators measure activities by man-days indicators. For each indicator, purpose, target, periodicity, responsible, scope and methods for calculating are defined. The ones which complete the dashboard are also identified.

Q.No	Country	Article	Ref. in National Report
72	Switzerland	Article 8.1	P, 63

Question/ Comment: Quality management system

Could you provide examples for the performance indicators used for measuring the effectiveness of ASN's actions?

Answer: For the regulation process, ASN measures the effectiveness of its actions for the Government when ASN is consulted on draft regulations. The performance indicator for these actions is the number of regulatory "opinions" given by the ASN to the Government on draft decrees and ministerial orders per year. The scope is defined in the Environment Code and the office of administration is responsible for the following of this indicator.
For the authorisation process, ASN measures its effectiveness on individual authorisations. For example, the authorisations in the medical field shall be delivered in 6 months from the reception of the authorisation file provided by the applicant. Authorisations are created on the information system so the data extractions from this system allow to put in evidence the number of authorisations delivered in 6 months.
Concerning the inspection process, performance indicators measure the number of inspections performed by ASN in nuclear safety, radiation protection and transport. An inspection program defines the number of inspections that shall be performed in each field. This

	allow ASN to have a graduated approach and measure the effectiveness of inspection and enforcement actions. Concerning the emergency preparedness and response process, performance indicators are used to measure the effectiveness of the ASN support in case of emergency situations. ASN indicators measure the acquittal rate of ASN agents during a test or realistic emergency alert.

Q.No	Country	Article	Ref. in National Report
73	Switzerland	Article 8.1	P, 63

Question/ Comment	Transmission of knowledge Could you describe ASN implementation of its kowledge-management program?
Answer	ASN has been implementing for many years a comprehensive training program for its staff. This program intends, in particular, to give ASN inspectors sufficient and required competency to perform their activities. This program is based on technical courses and also general courses in the field of communication, legal, quality, English language or management. Such training is taken into consideration in any decisions to qualify staff as inspectors. To maintain its training program updated and improve it if necessary, at the end of each year n a training program for the following year n+1 is set up according to and in line with performed and ongoing recruitments and activities to conduct. During the year n+1 this program is implemented, with or without amendments. Then, at the end of the year n+1, the implemented program is reviewed and compared with the planned program. This assessment gives ASN the possibility to improve its training offer and to complete or modify for the year n+2. If there is lack or a skill which seems deficient ASN has possibility to set up new training courses either with its own resources or by contracting with private or public companies.

Q.No	Country	Article	Ref. in National Report
74	Turkey	Article 8.1	63

Question/ Comment	Has ASN any extra budgetary revenue from fees and fines?
Answer	Since 2000, all the personnel and operating resources involved in the performance of the responsibilities entrusted to ASN have been covered by the State’s general budget. ASN gets no extra budgetary revenue from fees and fines except in very few cases. This was for example the case during the period 2010-2012 when ASN reviewed the safety options for a new pressurised water reactor known as ATMEA 1. This is also the case for some training activities performed by ASN on the behalf of the European Commission.

Q.No	Country	Article	Ref. in National Report
75	United Arab Emirates	Article 8.1	63

Question/ Comment	The National report states that in 2012, about 4,520 days of training were provided to the ASN staff. Could France give more details on the nature of the activities to develop and maintain the competence of the regulatory staff?
Answer	The training program implemented by ASN aims at giving ASN' staff a high level of competency and skills to conduct its missions. Each year, about more than 4 500 days of training for a pedagogical budget near to 550 M€ is given. In 2012, the days provided aimed to (i) improve development of individual technical competency (about 3 800 days), (ii) develop a common culture and facilitate professional evolution (about 260 days), (iii) reinforce ASN relations with its international partners (about 470 days) and (iv) facilitate communication with ASN publics (about 150 days). For example, in the field of individual technical competency courses were given in nuclear safety, radiation protection, radioactive substances transportation, labour inspection...

Q.No	Country	Article	Ref. in National Report
76	United Arab Emirates	Article 8.1	63

Question/ Comment	According to the National Report, ASN has defined performance indicators for measuring the effectiveness of ASN's actions as part of its quality management system. Could France elaborate on those indicators and how they are used in the management of the regulatory body?
Answer	The performance indicators are defined each year with a review of targets. They are described in a specific procedure of the management system documentation and this procedure is validated by the different heads and approved by the director general. The performance indicators are based on the generic processes of the management system. Processes pilots ensure the following of the performance indicators of their process. Each months, the indicators concerning inspection process are extracted from the information system in which inspections are created. These indicators are presented during a large executive committee meeting. They are used to follow the progress of the inspections program. Other indicators are extracted four times a year and forwarded to senior management through dashboards.

Q.No	Country	Article	Ref. in National Report
77	United Arab Emirates	Article 8.1	63

Question/ Comment	Please describe further the roles and interaction between IRSN as the major TSO and ASN in review and assessment of licensing submissions (eg licence applications, significant modifications, and periodic safety reviews). How does the regulatory body manage the relationship?
Answer	IRSN pools public expertise and research in the field of radiation protection and nuclear safety. So it works for ASN but not only. Considering expertise, ASN is still the main "customer" of IRSN. ASN pilots IRSN: - by giving an opinion on its budget, especially on the budget dedicated to the support of ASN ; - by establishing different framework documents : to clarify the objectives, methodology and means for each topic, including ethics considerations ;

- and by maintaining a permanent dialog to adjust the documents when necessary.
IRSN’s overall workforces stand at 1,700, about 500 of whom are devoted to ASN technical support.
Its total budget amounted to €315 millions, of which €84 millions were devoted to providing ASN with technical support.
ASN receives about 600 analyses from IRSN per year.
IRSN conducts about 30 analyses presented to advisory committees of experts to allow them to issue opinions and recommendations.
It can also be noted the number of co-ordinating meetings, about 60, and the number of technical meetings, very variable but around 500. Some of them are strategic. ASN is not the only participant. But its participation is necessary to be sure that its concerns are taken into account by the institute.
Some others are bilateral and directly oriented on the ASN/IRSN relationship.
In addition of these meetings, ASN and IRSN meet in seminars, either between entities to exchange on specific subjects in the daily relation, or, around more general topics as international cooperation and assistance, communication or human resources.
In this relation, ASN needs to pilot IRSN so that it gets relevant expertise for ASN making its decisions.
Therefore,
- ASN has to get a sufficient vision on the means engaged by IRSN ;
- ASN has to be able to have an opinion on the quality of the work performed by IRSN.

Q.No	Country	Article	Ref. in National Report
78	Japan	Article 8.2	p62

Question/ Comment French report says, □gno member may hold any interest such as to affect his or her independence or impartiality.□h
What provision or system do you have to assure independence of the ASN and IRSN staff, such as, limitation of redeployment to/from the promotion side?

Answer Resolution CODEP-CLG-2012-033820 of ASN Chairman of 4 July 2012 specifies the procedures for application within the ASN rules of conduct established by law n° 2011-2012 of 29 December 2012.
Persons concerned reveal links that each registrant currently has or had during the five years preceding his duties at the ASN, with legal entities whose activity falls within the competence field ASN safety of health products.
ASN is very committed to the independence of the work carried out by IRSN. It is particularly important that the expertise conducted on a particular issue is carried out by experts who have not contributed to its preparation.

Q.No	Country	Article	Ref. in National Report
79	Korea, Republic of	Article 8.2	66

Question/ Comment In page 66 of Article 8, it is stated that "These Committees(CLI) ....have a general duty to monitor, inform and discuss nuclear safety, radiation protection...". Please explain when(on which project phase) it is mandatory for regulator and operator to invite the Committees in terms of the regulation/project procedure.

Answer	The second paragraph of Article L.125-26 of the Environmental Code states that consultation of the CLI is required for any project that involved the public: DAC draft, Final shut-down and decommissioning draft, significant modification draft (art. 31 of Decree procedure). Pursuant to II of Art. 18 of Decree procedure, ASN consults CLI on draft regulations on water abstraction and effluent discharge.

Q.No	Country	Article	Ref. in National Report
80	China	Article 9	section9.2.1

Question/ Comment	Description In section9.2.1 Measures taken by EDF¡± As a responsible industrial firm and being aware of the particular nature of the nuclear power generating activity, EDF has always, since the beginning of operation of the NPPs, sought to inform the public about the operation of the facilities, technical events and activities concerning this form of energy in general and all safety aspects in particular¡± Question: How does EDF collect complete public feedback information after event? And how does EDF response to public feedback? How is the process?
Answer	1. EDF monitors public feedback on its activity using a range of complementary tools. The first tool is opinion polling. Quantitative and qualitative surveys are carried out every year at EDF's request by specialized polling institutes. The quantitative studies are conducted among local residents living within the range of 15 km around nuclear plants, both in operation and in decommissioning. They provide valuable insights regarding public perception of nuclear installations' safety level and the trust that local residents put in the nuclear operator. Moreover, every year, three or four nuclear power plants are selected to benefit from a qualitative survey conducted among a panel representative of all stakeholders involved. These studies allow for a better understanding of public expectations in terms of information and transparency policy. Other indicator that reveals the state of public opinion is information requests that citizens address to EDF. In compliance with the article L. 25-10 of The French Environmental Code, every person living near a nuclear installation can question the nuclear operator directly about any matter related to nuclear safety, radioprotection and environment. The operator has a legal obligation to provide answers within two months. Moreover, in compliance with ISO 14001 standards, every nuclear plant must register information requests and complaints expressed by the local population, with the view of providing specific and thorough answers. The local residents can also ask the operator for further information regarding its activity via Local Information Commissions (CLI). Established by law and set at every nuclear installation in France, these bodies gather all local stakeholders (local representatives, environmental associations, local industries etc). They function as a discussion platform and a reliable source of information for the local residents. 2. Every nuclear plant runs its own website that provides accurate and regularly up-dated information about safety, radiation and environment related events, as well as the results of measurements carried out in the surroundings to determine the level of radioactive and non-radioactive discharges. Moreover, every plant releases a monthly information letter, targeted at local stakeholders. It provides

accurate information about the station's activities and recent events. The letter also includes monthly records of radioactive and non-radioactive environmental discharges. In compliance with article L.125-15 and L 125-16 of the French Environmental Code, each power station releases an annual report that further develops topics related to safety, environment and radiation protection. Also, the Local Information Commissions' works as well as minutes from their meetings are available to the large public via Internet, on the Local Information Commissions' websites.

Q.No	Country	Article	Ref. in National Report
81	Luxembourg	Article 9	page 67

Question/ Comment

In a recent incident at the NPP Cattenom involving the accidental release of 58 cubic meter of 33% hydrochloric acid to the environment, the operatorâ€™s information to the public dealt with â€œa release corresponding to less than 1% of the annual discharge permits in accordance with operating proceduresâ€□. Knowing that prior to that release several rules and regulations were not applied, this type of information seems not in line with the EDF policy given under 9.2.1 of the national report. We would thus like to ask the following questions:

1. Is the EDFâ€™s policy as described an obligation resulting from the TSN act or is it voluntary?
2. What mechanisms are in place at EDF to ensure that this internal policy is respected?
3. Does ASN have powers to enforce the EDF policy?
4. May ASN promote the public information policies of the operators by of means? If yes, please explain.
5. To our knowledge, the TSN act deals with public information related to nuclear safety and radiation protection issues. Are there any plans to broaden that scope to other areas of public interest, such as incidents in the industrial parts of NPPâ€™s?

Answer

1) Order of 7 February 2012 setting the general rules relative to basic nuclear installations:

Article 2.6.4:

“The licensee notifies ASN of each significant event as soon as possible. The notification includes more particularly:

- the characterisation of the significant event;
- a description of the event and its chronology;
- the actual and potential consequence with respect to protection of the interests mentioned in article L. 593-1 of the environment code;
- the measures already taken or envisaged to deal with the event either provisionally or definitively.”

Article 4.4.1:

“In the event of accidental pollution originating within the bounds of the basic nuclear installation, the licensee immediately provides ASN, the Prefect, and the maritime Prefect if applicable, with all the necessary information for determining the measures to protect the interests mentioned in article L. 593-1 of the environment code that are threatened on account of this pollution.”

ASN Resolution 2013-DC-0360 of 16th July 2013 relative to control of nuisance effects and the impact of basic nuclear installations

on health and the environment:
Art. 5.3.1. - The report mentioned in article 4.4.4 of the abovementioned order of 7 February 2012, shall more specifically include the following information: […]
"- a list and brief description of the significant events entering into the field of application of this resolution and having formed the subject of a declaration in application of article 2.6.4 of the abovementioned order of 7 February 2012 and the corrective measures taken by the licensee;"
Individual regulations specific to each installation also include obligations of information of the regulatory authority and the public. Moreover, EDF produces shares information on a voluntary basis .

2) First of all, the release of hydrochloric acid into the environment that took place at the NPP Cattenom was not due to a failure in the implementations of rules and regulations but due to a default of the equipment for conditioning the cooling system. Following this event, the maintenance conditions have been modified to reinforce the liability of these equipments.
EDF's company policy is linked to its environmental system management. EDF company has obtained the 14001 ISO certification in 2002. Each NPP implements the principles of this policy in its proper policy and in its operational operating documents. This policy is known by all the people who intervene in the NPP, the EDF staff but also the subcontractors. The Operating nuclear Division has recently revised its policy, in order to integrate the recent obligations of BNI order (published in February 2012 and entered into force in July 2013), which demands to the Nuclear operators to elaborate and maintain an "integrated policy" which takes into account both safety issues and environmental protection in normal operation.

3) ASN is in charge of enforcing all applicable rules, including those mentioned in point 1). ASN controls the respect of declaration and information rules about significant events, like accidental pollutions.
Moreover, ASN can at anytime supplement new individual stipulations based on ASN resolutions.

4) The TSN Act in particular guarantees "the public's right to reliable and accessible information on nuclear security" (Article L.125-12 of the Environment Code). The right to information on nuclear safety and radiation protection concerns all fields of ASN activity, in particular:
– informing the public about events occurring in nuclear installations or during the transport of radioactive materials, about discharges or releases from nuclear installations;
– informing workers about their individual radiological exposure;
– informing patients about the medical procedure, in particular its radiological aspect.
ASN ensures application of these measures, which are binding on itself as well as on the licensees subject to its regulation, and the implementation of which can sometimes lead to confusion. ASN attempts to facilitate the exchanges between all the stakeholders

concerning the problems encountered and best practices.

All nuclear installation licensees must therefore establish an annual report on their situation and the steps they take with respect to nuclear safety and radiation protection.
In 2010, after extensive consultation, in particular with the CLIs, ASN published a guide on the drafting of these reports on www.asn.fr, so that they meet the goals of the Act and deliver the most complete and accessible information possible to the general public. This guide recommends that the reports not be limited to simple application of the letter of the law, but that they give a broader picture of the impact of the facilities and the steps taken to reduce the risks of accidents and chronic detrimental effects.
Every year, ASN analyses the reports, not simply verifying compliance with the letter of the law, but also aiming for continuous improvement in the quality of the information distributed to the public. Its assessments are realised each year in ASN's annual report.

5) The provisions of TSN Act on Public Information Act apply to any activity in the INB perimeter of a nuclear power plant.

Q.No	Country	Article	Ref. in National Report
82	Belgium	Article 10	chap. E, sec. 10.4, page 73

Question/ Comment -
Little information seems to be given in §10.4 on the assessment by ASN of the safety culture programmes and their development.
What methodes and techniques are used by ASN to independently monitor the safety culture of the licensees?
What process will these independent safety culture observations and conclusions thereof follow?

Answer The assessment of ASN focuses on safety management practices instead of measuring safety culture. The topics of the safety management inspections performed by ASN are for example : 1. The competencies management system : ASN asks the licensee to have a rigorous competencies management system, including determination of needs, both in respect of manpower qualification and numbers, and a programme of specialized training and qualification through experience. ASN monitors the competencies management system for all operators working on the facilities (internal and external workers). 2. The organisation of the licensee to operate experience feedback: ASN monitors how NPPs are organized to analyze deviations and significant events, the methodology used, the depth of analyzes, and the development and implementation of the outcomes of these analyzes.

Q.No	Country	Article	Ref. in National Report
83	Belgium	Article 10	chap. E, 10.3.1 page 71-72

Question/ Comment -
In what respect has the three-year plan been adapted to take into account lessons learned from Fukushima?

Answer The three-year plan on safety policy was already adapted by the end of 2011 to take into account lessons learned from Fukushima. It includes for 2012-2104 a specific item on the implementation of stress-test studies, known as ECS (complementary safety

	assessments). The 36 nuclear facilities of CEA have been prioritized in three batches, according to the likeliness of finding cliff-edge effects. The first priority batch was already processed by the end of 2011. The ECS studies of the second priority batch are being delivered according to the ongoing plan.

Q.No	Country	Article	Ref. in National Report
84	Canada	Article 10	Page 69, Section 10.1, Paragraph 4 and s

Question/ Comment	Can you clarify whether the resolution and the guide to safety policy and management in the BNIs (Guide no. 15) referred to in Section 10.1 are expected to be published and how this may impact on the Nuclear Safety Policy adopted by EDF Group in 2012?
Answer	The resolution on safety policy and integrated management system has been submitted for public consultation on the ASN website from 24 January until 21 February; it should be published by the end of the first half of 2014. The publication of the guide on safety policy and integrated management system is expected within the next two years. It is too early to have a definitive opinion of the impact of those documents on EDF's Nuclear safety policy. Anyhow, it is underlined that, according to the ministerial of 7th February 2012, EDF has to analyse its policy at least every 5 years and update it if necessary.

Q.No	Country	Article	Ref. in National Report
85	Luxembourg	Article 10	page 70

Question/ Comment	The report states that EDF has in the past three years aimed to further reinforce the safety culture of each party. Also in 2012, the EDF Group adopted a Nuclear Safety Policy that gives further guidance on safety culture. In this relation, we have the following questions: 1. Has there been any clear lack of safety culture identified as a trigger for this more intensive focus on safety culture? 2. Does EDF monitor the effect of this reinforced focus on safety culture? Please also report conclusions if available. 3. Has ASN observed any positive development in EDFâ€™s safety culture approach over those 3 years? Is there for example a positive trend concerning human errors leading to incidents?
Answer	1) No clear or specific lack of safety culture has been identified as a trigger for this policy. The new safety policy is the ""EDF group safety policy"". It includes safety fundamentals of the last safety policy EDF SA such as priority given to safety, development of safety culture, research of excellence. The new policy is more precise on how to implement and formalize our major practices. 2) EDF SA conducts analyzes, assessments and annual safety reviews as part of its integrated management system levels that assess progress, identify weaknesses and areas for improvement and actions consistent with objectives defined by the policy. These reviews also help if necessary for re-examining the content of the policy. 3. The assessment of ASN focuses on safety management practices instead of measuring safety culture. Regarding operating experience feedback, one of the tendencies over the last three years is the significant renewal of EDF's workforce that lead to an increased proportion of due to insufficient experienced people. On the other hand, ASN notes that EDF made significant effort to improve the operating experience feedback process. Overall, the number of significant related errors, that could have been avoided with effective implementation practices reliability

activities, is in steady decline (around 60% since 2007).

Q.No	Country	Article	Ref. in National Report
86	Romania	Article 10	10.4.2

Question/ Comment
On page 74 of the report, reference is made to "an internal safety check body, independent of the operational side and structured at several levels". Is this an organizational unit inside licensees' organizational charts that fulfills the role of "nuclear safety oversight" or "internal regulator" as referred to in other jurisdictions? Please provide more information about this "independent safety function" with regard to the following:

- is it provided for each site or it is set up at corporate level for all the sites?
- is it established voluntarily by the licensees or is it required by ASN through regulations or license conditions?
- what are the competences required for the staff performing this independent safety function and to whom do they report in the licensee's organisation?

-what is the interaction between the licensees' staff performing the independent safety function and the ASN inspectors?

Answer
An in-house independent nuclear assessment function is put in place at every level of the organisation. Each function reports independently of all lines functions and has not only the right, but also the duty, to notify senior management of inappropriate or inadequate line management process. A multilevel organization is established within the Company to provide independent monitoring and assessment of nuclear safety. Safety oversight and monitoring functions are established at each level of the Company. At each level, responsibilities rely on line management. Opposite the line management, the “Nuclear Safety Independent Line” brings a double support : verification and assessment & support and advice to the line management. Specific oversight committees are in place at each level to share / confront points of views and decide actions. The chairmen of these different committees are at each management line level. To be accredited, the Safety Engineer on site shall pass an examining board from the corporate level. At the nuclear generation division corporate level, the Nuclear Safety Director challenges the Nuclear Operations Division Director ; he also supports and advices him. He is supported by the Nuclear Inspectorate that performs overall safety assessments at the nuclear power stations. At the GENCO level, the General Nuclear Inspectorate for Nuclear Safety ensures that the Group meets all nuclear safety and radiation protection requirements.

Generally speaking, the interactions between the plant and ASN’s inspectors goes through the management of the plant. It is not provided for that the licensees' staff performing the independent safety function reports to ASN’s inspectors. ASN’s performs regularly inspections dedicated to the independent safety function.

Q.No	Country	Article	Ref. in National Report
87	Romania	Article 10	10.4.2

Question/ Comment
How does ASN review and assess the nuclear safety culture of the licensees and to what extent? What are the information and indicators considered by ASN as relevant to the nuclear safety culture of the licensees? To what extent are the reviews and inspections

	performed by ASN on licensees' management systems provide information relevant to nuclear safety culture?
Answer	The assessment of ASN focuses on safety management practices instead of measuring safety culture. The topics of the safety management inspections performed by ASN are for example : 1. The competencies management system : ASN asks the licensee to have a rigorous competencies management system, including determination of needs, both in respect of manpower qualification and numbers, and a programme of specialized training and qualification through experience. ASN monitors the competencies management system for all operators working on the facilities (internal and external workers). 2. The organisation of the licensee to operate experience feedback: ASN monitors how NPPs are organized to analyze deviations and significant events, the methodology used, the depth of analyzes, and the development and implementation of the outcomes of these analyzes.

Q.No	Country	Article	Ref. in National Report
88	Switzerland	Article 10	10.1, P. 69

Question/ Comment	The report states that “historically, this safety management system is based on the development of a nuclear safety culture”.
	Please, would you outline the relationship between safety management system and safety culture that is addressed in the above mentioned sentence.
Answer	The objective of the safety management system is to ensure that the requirements relative to safety are always taken into account in any decision concerning the installation. It is then linked with safety culture.

Q.No	Country	Article	Ref. in National Report
89	Switzerland	Article 10	10.1, P. 69

Question/ Comment	The report states that ASN will be publishing a resolution and a guide to safety policy and management in the BNIs (Guide No. 15).
	When will this guide be available for the public?
Answer	The resolution on safety policy and integrated management system has been submitted for public consultation on the ASN website from 24 January until 21 February; it should be published by the end of the first half of 2014. The publication of the guide on safety policy and integrated management system is expected within the next two years.

Q.No	Country	Article	Ref. in National Report
90	Ukraine	Article 10	para 10.2

Question/ Comment	«Thus, on the basis of the system built up gradually since the beginnings of the nuclear fleet, EDF has in the past three years aimed to further reinforce the safety culture of each party»: What indicators does EDF apply for assessment of the level of safety culture?
Answer	Several indicators used by EDF light up indirectly the safety culture aspects (for example : Tech. spec. non-compliances, Non-

	compliance circuit configuration, Automatic reactor scrams…). The analysis of these indicators in the frame of the in-depth analysis of events provides information on the functioning of organizations and practices in the field. In addition, each site, in its annual safety analysis, generates diagnosis on safety management, particularly regarding in the 3 key principles : safety leadership, personal development and commitment, oversight and continuous improvement. Finally, every 2 years, nuclear inspectorate establishes an evaluation based on interviews and field observations.

Q.No	Country	Article	Ref. in National Report
91	United Arab Emirates	Article 10	74

Question/ Comment	The National Report states that ASN sees the presence of an internal safety check body within the licensee, independent of the operational side, as a strong point. This independent body is said to benefit from significant resources as well as clear support from the management of the NPPs. However, the National Report also refers to problems with filling certain safety engineer positions and a trend towards these positions being occupied mainly by young safety engineers from the operations sections. It further says that this trend could lead to the independent safety function experiencing problems in the complete performance of its duties with respect to the operations side. Could France give some details on the role and function of the internal safety check body and the potential problems that are referred to?
Answer	An in-house independent nuclear assessment function is put in place at every level of the organisation. Each function reports independently of all lines functions and has not only the right, but also the duty, to notify senior management of inappropriate or inadequate line management process. A multilevel organization is established within the Company to provide independent monitoring and assessment of nuclear safety. Safety oversight and monitoring functions are established at each level of the Company. At each level, responsibilities rely on line management. Opposite the line management, the “Nuclear Safety Independent Line” brings a double support : verification and assessment & support and advice to the line management. Specific oversight committees are in place at each level to share / confront points of views and decide actions.. The number of people and skills requirements of the personal independent safety line are established and monitored at nuclear division level, an annual diagnosis is required at each site to assess the robustness and listening to its independent safety, this point is also periodically evaluated by the nuclear inspectorate. The safety engineer shall pass an examining board from the corporate. In case of potential cyclical problems in filling such positions, cross-movements of qualified safety engineers between sites can be implemented. The nuclear plants are survey by an independent body that looks at the day to day work and at the medium an long term work. The survey of the day to day work is based on the confrontation of “haw it is done” and “how it should be done” The medium and long term works look at the process, the check lists and how it is implemented and for which results.

	The regulatory body, during inspections, noticed that the skills and resources involved in this independent body are not always in accordance with these tasks. These are tendencies that should give rise to particular attention by the operator and a specific monitoring by the regulatory body.		
Q.No 92	Country United Arab Emirates	Article Article 10	Ref. in National Report 70
Question/ Comment	Could France give some examples of measures (good practices) that have enhanced safety culture at the NPPs and at the regulatory body?		
Answer	Example of Good practices used to enhance safety culture : Human performance program, human performance tools and managers on the field, Annual safety analysis, New employees and managers training , leadership program, Decision makings tools, Use of Safety management guide for manager and self assessment.		
Q.No 93	Country United States of America	Article Article 10	Ref. in National Report 10.4.1
Question/ Comment	The report states that an IRRS follow-up mission occurred in 2009. During that follow-up, the IRRS team determined that 90% of the recommendations made in 2006 were implemented. The IRRS team also determined that there were a few areas for improvement, notably in skills management. Please describe the actions taken since 2009 to improve skills management.		
Answer	Many actions have been taken since 2009 and 2010 in the field of skills management. Firstly, to better anticipate the need for its own staff each manager is involved in setting up provisional training program. At the end of each year a training program for the next year is set up according to and in line with performed and ongoing recruitments and activities to conduct. During the year this program is implemented, with or without amendments. Then, at the end of the year, the implemented program is reviewed and compared with the provisional program. This assessment gives ASN the possibility to improve its training offer and to complete or modify for the next year. If there is lack or a skill which seem deficient ASN has possibility to set up new training courses either with its own resources or by contracting with private or public companies. Secondly, ASN implements on a permanent way new courses that have been identified necessary. This was the case for example in 2012 and 2013 with new courses in the field of regulation, of environment, of metallurgy, of management, of human factors. On a similar way improvements are given to courses if necessary either in their content or in the general organisation. Thirdly, ASN conducted in 2013 a complete review of its training program for qualifying its inspectors which had been implemented for years. This review consisted in optimization of the different modules which are mandatory to be qualified as inspector with the aim to give the right (ie which meets expectations and needs in competency) courses at the correct time (ie just at the right time).		
Q.No 94	Country Romania	Article Article 11.2	Ref. in National Report 11.2.4 ASN analysis and oversight

Question/ Comment	What review and inspection guidelines, criteria and indicators does the ASN staff use to the assess the sufficiency and adequacy of the licensees' human resources?
Answer	The ministerial order of 7th February 2012 requires that the licensee specify, in its management system, the organization and the resources to mobilize for managing the risk prevention and inconveniences. This management system has to be reassessed periodically by the licensee. ASN regularly analyses licensee's competencies management system generally speaking or focusing on specific activities (for example : for maintenance activities, control room operators activities, oversight of external contractors...); to do so, ASN can request specific assessments to IRSN and/or the Advisory Committee for nuclear reactors (GPR) and taking into account the experience feedback of the significant events notified by the licensee, especially by analysing the real underlying causes of those events. In inspection, ASN verifies that the licensee comply with its safety management system, especially in terms of organization and resources, and has established a cartography of the skills/competencies and the necessary resources as defined in its management system.

Q.No	Country	Article	Ref. in National Report
95	Switzerland	Article 11.2	11.2.1, P. 78

Question/ Comment	The report states that the volume of training at the NPPs has risen significantly over the past 5 years. What are the reasons for this significant increase of training hours?
Answer	Adapting and renewing the skills of NPP's personnel is one of the main stakes for EDF Nuclear Generation. The reason is that more and more employees are getting retired since 2006. In this context, EDF Nuclear Generation has decided to face this issue with anticipation: This has resulted in the recruitment of a large number of new employees. A variety of them are coming from EDF Units that have been reorganized while others come from tertiary departments where less human resources are needed. As a conclusion, the number of training courses is in sharp rise mainly because of the substantial increase in the number of initial training programmes.

Q.No	Country	Article	Ref. in National Report
96	Switzerland	Article 11.2	11.2.4, P. 79

Question/ Comment	The report states that human resources are checked during BNI decommissioning inspections. What kind of human resource topics are inspected during BNI decommissioning inspections?
Answer	Human resources regulatory requirements applicable to decommissioning are the same as for any other stage in the life-cycle of the facility. Each item of the regulation can be subject to an inspection. However some of them are more relevant during the decommissioning phase than during other phases of the life-cycle of the facility. The ministerial order of 7th February 2012 (BNI order) states that " Implementation of the post-operational clean-out and dismantling methods and techniques takes into consideration the organisational and human factors to determine the conditions for safe and effective performance of the activities and prevent the

risks of inappropriate actions." Organisational and human factors shall be taken into account in the application file for decommissioning and in the review by the authority leading to the authorisation. They can then be subject to a regulatory control of that basis. Topics that can be inspected can be for example the training and qualifications of workers (BNI order art. 2.5.5 and in the decommissioning plan to be issued by the operator during the lifetime of the facility where the operator has to describe " Provisions made by the operator for maintaining skills and knowledge of the facility", see guide ASN n° 6, appendix 1, point B.4).
ASN pays also attention to issues that become particularly relevant during the decommissioning phase: radiological protection of workers and the general public (Labour Code art. R. 4451-1 to R. 4451-130 and Public Health Code art. R.1333-1 to R.1333-12), monitoring of outside contractors (ministerial order of 7th February 2012, art. 2.2.1 to 2.2.4).

Q.No	Country	Article	Ref. in National Report
97	United Arab Emirates	Article 11.2	79

Question/ Comment: The National Report ASN checked personnel training for severe accident management during targeted inspections in 2011. Could France elaborate on the scope and content of such training provided to licensee staff, and also what training in this area ASN conducts for its staff?

Answer: The French regulations and the EDF on-site emergency plans (PUI) requires that severe accident drills should be performed at regular intervals. Each nuclear power plant must carry out several drills each year, including one in which the on-site emergency plan is deployed. The number of drills per year and per site is determined according to the number of emergency team members, as each team member must attend one PUI exercise per year.
As a conclusion of the stress tests performed after Fukushima Daiichi accident, ASN required the licensee to provide the personnel concerned with the training and preparation needed to enable them to respond to particularly stressful accident situations. It shall ensure that the outside contractors liable to intervene in management of the emergency adopt similar requirements concerning the preparedness and training of their own staff.
Each member of ASN staff receive training in emergency management. Each staff member participates regularly in emergency drills and is brought to hold several positions in the emergency centre team.

Q.No	Country	Article	Ref. in National Report
98	United Arab Emirates	Article 11.2	77

Question/ Comment: Could France elaborate on the criteria applied by the regulatory authority to assess sufficiency of staff resources at the NPPs?

Answer: The ministerial order of 7th February 2012 requires that the licensee specify, in its management system, the organization and the resources to mobilize for managing the risk prevention and inconveniences. This management system has to be reassessed periodically by the licensee. ASN regularly analyses licensee's competencies management system generally speaking or focusing on specific activities (for example : for maintenance activities, control room operators activities, oversight of external contractors...); to do so,

	ASN can request specific assessments to IRSN and/or the Advisory Committee for nuclear reactors (GPR) and taking into account the experience feedback of the significant events notified by the licensee, especially by analysing the real underlying causes of those events. In inspection, ASN verifies that the licensee complies with its safety management system, especially in terms of organization and resources, and has established a cartography of the skills/competencies and the necessary resources as defined in its management system.

Q.No	Country	Article	Ref. in National Report
99	Belgium	Article 12	chap. E, 12.4.2 page 82

Question/ Comment	- Could you develop further on the integration of organisational and human factors into the design process for the JHR. What are the main lessons learned from this integration?
Answer	According to the order of 7 February 2012 setting the general rules relative to BNI, the nuclear safety demonstration shall integrate the technical, organisational and human dimensions. For the JHR : organisational and human factors (OHF) were taken into account in the definition of safety options by CEA in a imperfect way : the consideration of this factors for the design was entirely delegated to the CEA prime contractor. In this sense, the ASN requested a detail action plan which should be drawn up on the entire field of OHF, clearly defining the objectives and principles of safety and specifying the coordination between the operator and the designer. This plan was provided and must be stated in support of the request for commissioning the installation. Integrating OHF led, by example at the separation of two distinct work areas (operations and experiments) in the reactor building. The feedback from other facilities highlights indeed a problem in terms of radiation protection during transfer operations. Instead, an ergonomic assessment led to design the control room of the experiments near the control room of the facility to facilitate communication.

Q.No	Country	Article	Ref. in National Report
100	Canada	Article 12	Page 80, Section 12.4.2

Question/ Comment	The report discusses the integration of organizational and human factors as pertains to safety. However, no mention is made of operational human performance programs, corrective action programs nor of the use of human factors engineering knowledge and processes in a proactive manner to ensure operability, maintainability and safety. Can ASN elaborate on the integration of these programs?
Answer	The licensee has developed a methodology to transform the engineering practices, in order to take into account the people needs in the development of systems and modification of materials and organizations (cf. User-centred design). ASN considers the philosophy of this methodology relevant and important to ensure the safety of installations and safety of workers. The licensee has developed a Corrective Action Program which aims to analyse and correct all the deviations. The licensee has also developed a Human performance program which comprises intervention reliability enhancement practices: pre-job, stop and think, self-check, cross-check, secure communication, debriefing.

Q.No	Country	Article	Ref. in National Report
101	Japan	Article 12	p80

Question/ Comment

French report says, □gASN is counting on integration of OHF, in the following fields of activity: the activities carried out for operation throughout their service life□h in page80, and □gThe major work themes identified are: - management of emergency situations□h in page 82.
Are emergency situations included in the service life?
What kind of OHF requirements are imposed for emergency situations?

Answer

After the Stress tests, ASN defined requirements. The licensee is expected to: 1) define the human actions required for management of the extreme situations studied in the complementary safety assessments (CSAs). It shall check that these actions can effectively be carried out given the intervention conditions likely to be encountered in such scenarios. It shall for instance take account of the relief of the emergency teams and the logistics necessary for the interventions. It shall specify any material or organisational adaptations envisaged; 2) send ASN a list of the necessary emergency management skills, specifying whether these skills rely on external contractors. The licensee shall provide proof that its organisation ensures the availability of the necessary skills in an emergency situation, including if external contractors are used.
3) provide the personnel concerned with the training and preparation needed to be able to respond to a particularly stressful accident situation. It shall ensure that the external contractors who could have to intervene in management of the emergency adopt similar requirements concerning the preparedness and training of their own staff. 4) define the social and psychological care to be provided for the emergency teams, taking account of the family environment, implemented in a particularly stressful accident situation, to ensure working conditions allowing emergency management that is as effective as possible (see on the ASN website the resolution of 26 June 2012, [EDF-CAT-26][ECS-35]).

Q.No	Country	Article	Ref. in National Report
102	Korea, Republic of	Article 12	2, page 80

Question/ Comment

According to the description of Section 12.2, the organization set up by EDF makes provision for a "human factors consultant" position per pair of reactors. Please explain how many staffs with human factors expertise are working in the utility and the regulatory body, and what kinds of duties they perform in their organization.

Answer

The position of Human Factors Consultants (HFC) was established since 1993 at each site and a team of experts has existed since the mid-80s at the corporate level and Research & Development. Since 2004, each engineering centre also has an HF expert. In 2014, there are more than 30 site HFC, 6 national experts and a team of 10 persons in R&D. The major activities of site HFC include : - Operating Experience, with the completion of 2nd level analysis and qualitative analysis on field to investigate of weak signals (from the Corrective Action Programme approach, as established between sites and corporate level), - The appropriation of approaches and tools by operators/workers on all areas, with priority given to the development of safety culture, - Training for Human Performance

	approaches and tools. On departments or teams request, HFC can intervene to make a diagnosis and to propose or co-build actions for improvement.

Q.No	Country	Article	Ref. in National Report
103	Pakistan	Article 12	General

Question/ Comment	France may like to share information regarding typical errors that have been observed after analysis of the events connected with human activity and organizational factors.
Answer	First example of typical explanations after analysis of the events connected with human activity and organizational factors: The intervention required the locking of a valve. This valve was not directly reachable and must be physically manipulated remotely by means of a pole. Furthermore, the control of the valve position was not visible from where the worker stood to manipulate the valve. Finally, in order to open or close the valve, the worker has to turn the wheel 80 times. In this context, the probability of a positioning error of the valve was significant. Second example : One control room operator had to perform measurements with a stable power of the reactor for at least 6 hours. To keep the stability of the reactor power, the operator performs dilutions every 15 min. During each dilution, the operator must carry out ten successive actions on the control panel in a limited time. The control panel wasn't well designed : the lack of visibility of water gauge and the repetition every 15 min of this sequence could lead the operator to make an error. In addition, many other activities were carried out by the operator at the same time. This operator was also bothered by a specific request on the turbine and thus has forgotten the activity dilution. The dilution was automatically stopped after reaching a critical threshold.

Q.No	Country	Article	Ref. in National Report
104	Romania	Article 12	12.4.1

Question/ Comment	It is mentioned in the report that "With regard to safety management more broadly, ASN observes that there is now greater presence of EDF managers in the field, mainly to disseminate and implement managerial policies and requirements. ASN considers that this presence however still does not contribute enough to a better understanding of the realities in the field on the part of the site’s management." What are the regulatory expectations of ASN regarding this matter?
Answer	ASN's expectations regarding this matter are that field visits should be a tool to make the worker activities easier by : - improving working conditions (e.g. notice the good practices) in order to allow the workers to use their competencies; - reminding the expectations to the workers.

Q.No	Country	Article	Ref. in National Report
105	Romania	Article 12	12.2

Question/	The use of INSAG-18 is mentioned as regards the licensees' processes for the management of organizational change. Does the ASN

Comment	review and approve organizational changes proposed by the licensees? If yes, what categories of changes need to be submitted for regulatory review and approval and what are the regulatory guidelines and criteria used for the review and approval of organizational changes?
Answer	The ministerial order of 7th February 2012 incorporates requirements to specify, in the integrated management system of the licensee, the implemented provisions in terms of organization and resources to assure the risks prevention and inconveniences. ASN controls the effective application of these specifications in inspection and assesses, with the support of the IRSN and of the permanent group of expert, the effects of the changes of organizations on the level of prevention of these risks, mostly after their implementation. Only the organizational changes which have an effect on the organization defined in the general operation rules - RGE - (for example changes in details of implementation of subcontractor's supervision) may need an agreement from the ASN before their implementation. ASN is currently developing a new regulatory which will clarify the contents of the RGE regarding organization.

Q.No	Country	Article	Ref. in National Report
106	Switzerland	Article 12	12.4.1, P, 81

Question/ Comment	The report states that ASN considers that the presence of the management in the field still does not contribute enough to a better understanding of the realities in the field on the part of the site’s management. In article 10 (chapter 10.2 / page 70) it is stated that at EDF safety culture is promoted by several main leadership attitudes. One of these attitudes is “priority for safety is the role of each manager”. Then the report further states that the implementation of this attitude takes place within the deployment of a human performance project, which foresees measures like “analysis of field visits”. What are the criteria for the analysis of field visits conducted by managers in the above cited context? Is the understanding of the realities in the field of the site’s managers also one of these criteria? Does there exist a regulatory basis for field visits?
Answer	Elements of answer from EDF : Field visits are prepared by all managers of a department in order to cover the majority of activities and a minima the most sensitive, or those pointed by the Operating Experience as weak. Then, the managers prepare the visit by consulting the documentation relating to the activity and standards that the site produced. After completion of visit, further the exchange with the participants, managers collect their positive and negative findings in a database. They have a descriptive categorization, which includes a list of lines of defence "technical, human, organizational". In conclusion, there is no a priori criteria for the analysis of field visit, but during the preparation, the managers identify the expected features for the observed activity, the desired attitudes ; concerning the prevention human errors practices, there is a more precise reference. ASN's expectations regarding this matter are that field visits should be a tool to make the worker activities easier by : - improving working conditions (e.g. notice the good practices) in order to allow the workers to use their competencies - reminding the expectations to the workers.

Q.No	Country	Article	Ref. in National Report
107	Belgium	Article 13	chap. E, sec. 13.2, page 84 and sec. 13

Question/ Comment

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§13.2 of the French National Report provides some information on QA provisions at EDF with respect to work performed by contractors but it is not clear if these provisions apply as well to vendors and suppliers of safety related systems, structures and components (SSC). Please elaborate on the QA provisions in place at EDF with respect to vendors and suppliers of safety related SSC. Again §13.4.1 mentions only regulatory oversight practices on QA with respect to use of contractors at EDF. §13.4.1.2 mentions explicitly maintenance work performed by contractors and §13.4.1.3 mentions also inspection of engineering activities. To what extent is this oversight also applied to procurement of safety related SSC? Please elaborate on the nature of regulatory oversight performed by ASN on QA provisions with respect to procurement of SCC.
To what extent have policies and practices with regard to vendor and supplier qualification and oversight evolved in the light of the recent construction in France of new nuclear power plants (both at EDF and ASN)? What was the basis for these changes?

Answer

QA provisions are now set by the Ministerial Order of February 7, 2012. The basic expectation is that, once identified as an item important to safety (either safety related or safety item), explicit requirements will be set so that the item fulfil its safety mission, including under accident conditions if needed, as postulated in the safety analysis report. The fulfilment of each explicit requirement is to be checked.
If the licensee does not manufacture, operate or maintain the item and contracts such operation to a vendor/supplier/service provider, the licensee has to implement a supervision program to ensure the vendor/supplier/service provider has set up an effective system to have all explicit requirements relevant to fulfil its task.
Inspections performed by ASN at suppliers workshop enable ASN to verify whether explicit requirement have actually been met and whether EDF supervision process is effective.

QA provisions implemented by EDF regarding vendors and suppliers of safety related SSC are done on a multi level process

The assessment process of suppliers starts before purchase activities, when EDF evaluates potential bidders against technical and QA (such as ISO 9001) requirements. So the global purchase process aims at allocating the contract to the ""best bidder"" but also permits to qualify vendors as ""EDF possible suppliers"".

During completion of purchase activities, manufacturing Quality evaluation is performed by EDF, which provide an in-factory assessment of the bidders.

After the completion of the contract activities performed in Nuclear Power Plants, a contract performance assessment is done by EDF and presented to the suppliers through a feedback process. For in-factory activities, the same assessment process is performed on an annual basis.

Then every year EDF updates its vendors qualification list regarding QA provision and the completion of the suppliers design, manufacturing and on site activities as a mean to improve its operation experience feedback.

New ""Social Specifications"" as listed in § 13.2 complemented former rules also dedicated to vendors and suppliers."

Q.No	Country	Article	Ref. in National Report
108	Canada	Article 13	Section 13.4

Question/ Comment: Has ASN conducted safety culture evaluations at the Flamanville site, or of other licensees, vendors, or contractors? How frequently does ASN review safety culture of operating nuclear power plants, or of its own regulatory oversight programs?

Answer: The assessment of ASN focuses on safety management practices instead of measuring safety culture. During the inspection performed in Flamanville3, ASN pays special attention to safety management, for example the training of the teams in charge of the future operating and the safety knowledge of the construction team of the reactor. Thanks to the inspection findings, ASN can have a point of view concerning the management safety of the licensee. Concerning safety management of the subcontractors, ASN checks during the inspection in the suppliers workshop that EDF oversight takes into account the safety management.

Q.No	Country	Article	Ref. in National Report
109	China	Article 13	section 13.2

Question/ Comment: CHAPTER 13.2 Relations with contractors
¡± These social specifications and the proposals for regulatory changes were sent to the Prime Minister, the Minister for Industrial Renewal and the Minister for Ecology on 20th July 2012. It was also presented to the Steering Committee for Social Organizational and Human Factors, chaired by ASN, on9th November 2012.
Implementation of these social specifications in the contracts will begin in financial year 2013.¡±
Question:
What are the details of social specifications?
What¡¯s the result of these social specifications implementation in 2013?

Answer: The social specifications are now part of the expectations required for each contractor working with EDF. The initial assessment to be able to work for EDF and the renewal assessment to maintain this certification takes into account the social specifications. Part of the contractors expectations is his capacity to drive and control his subcontractors according to the EDF expectations. When working in the plant, the contractors management of his subcontractors is also assessed. The social specifications give, to all the contractors and subcontractors, expectations and responsibilities contributing to safety culture which makes a big step with other part of the industry .

The social specifications are expectations common to all actors in the field, designed to guarantee know-how, skills, training and qualification, as well as the adoption of nuclear safety, radiation protection, occupational risk prevention and quality of life at work as fundamental and essential criteria. The implementation of the social specifications is controlled with specific assessments, at the initial assessment of the contractors to get certification, at the renewal assessment of the contractors to maintain certification, and each time when working on the plant. As the social specifications are relatively new and in progress for implementation, a specific steering committee is in place to follow this initial phase.

Q.No	Country	Article	Ref. in National Report
110	China	Article 13	section13.2

Question/ Comment Description In section13.2¡± verifications, with the necessary distance and independence, to ensure correct implementation of the quality requirements...¡±
Question:
How to keep the necessary distance and independence?

Answer An in-house independent nuclear assessment function is put in place at every level of the organisation. Each function reports independently of all lines functions and has not only the right, but also the duty, to notify senior management of inappropriate or inadequate line management process. A multilevel organization is established within the Company to provide independent monitoring and assessment of nuclear safety. Safety oversight and monitoring functions are established at each level of the Company. At each level, responsibilities rely on line management. Opposite the line management, the “Nuclear Safety Independent Line” brings a double support : verification and assessment & support and advice to the line management. Specific oversight committees are in place at each level to share / confront points of views and decide actions. The chairmen of these different committees are at each management line level. To be accredited, the Safety Engineer on site shall pass an examining board from the corporate level. At the nuclear generation division corporate level, the Nuclear Safety Director challenges the Nuclear Operations Division Director ; he also supports and advices him. He is supported by the Nuclear Inspectorate that performs overall safety assessments at the nuclear power stations. At the GENCO level, the General Nuclear Inspectorate for Nuclear Safety ensures that the Group meets all nuclear safety and radiation protection requirements.
Generally speaking, the interactions between the plant and ASN’s inspectors goes through the management of the plant. It is not provided for that the licensees' staff performing the independent safety function reports to ASN’s inspectors. ASN’s performs regularly inspections dedicated to the independent safety function.

Q.No	Country	Article	Ref. in National Report
111	Japan	Article 13	p86

Question/ Comment French report says, □gASN conducts a number of inspections every year at suppliers of the reactors.□h
What inspections does ASN conduct directly at subcontractors as well as at reactor suppliers?

Answer	For FLA3 reactor under construction, ASN performs inspections in the suppliers workshops. The main objective of these inspections is to ensure that EDF's oversight of its supplier and of the potential subcontractors of its supplier is adequate. In 2013, 2 inspections were performed in suppliers workshop: the first one was dedicated to the ex-core instrumentation and the other to the containment heat removal system strainers located at the bottom of the reactor building. ASN also performs inspections during the manufacturing of nuclear under pressure equipments (like steam generators or the reactor pressure vessel). Those inspections aim to ensure that the equipment manufacturer takes into account hazards due to energy contained in the equipment but also specific requirements such as radioactivity or safety for instance. Because of the internationalization of nuclear component manufacturing, a careful monitoring of subcontractors is needed. The majority of those inspections are made by third party bodies mandated by ASN. In 2013, ASN carried out 15 inspections of manufacturers and their subcontractors, and 15 inspections for the surveillance of third party bodies. Third party bodies carried out in 2013 around 2700 inspections related to the conception and the manufacturing of equipment for the Flamanville 3 EPR reactor and around 2000 inspections related to the conception and the manufacturing of steam generators for the in service reactors.

Q.No	Country	Article	Ref. in National Report
112	Korea, Republic of	Article 13	4.1.1, page 86

Question/ Comment	It is stated in Clause 13.4.1.1 that ¡°During its inspections on sites under construction or already in operation, regardless of the field to be checked, ASN verifies that the quality assurance principles are respected. The adequacy of resources for tasks, etc. ...¡± 1) Is ASN responsible for QA auditing of EDF, and/or its contractors? 2) In order to evaluate the adequacy of licensee¡¯s resources for tasks, there must be regulatory requirements. Is there any regulation promulgated by the ASN or regulatory position regarding resources for tasks? How does ASN evaluate the adequacy of licensee¡¯s resources for tasks?
Answer	The ministerial order of 7th February 2012 requires that the licensee specify, in its management system, the organization and the resources to mobilize for managing the risk prevention and inconveniences. This management system has to be reassessed periodically by the licensee. ASN regularly analyses licensee's competencies management system generally speaking or focusing on specific activities (for example : for maintenance activities, control room operators activities, oversight of external contractors...); to do so, ASN can request specific assessments to IRSN and/or the Advisory Committee for nuclear reactors (GPR) and taking into account the experience feedback of the significant events notified by the licensee, especially by analysing the real underlying causes of those events. In inspection, ASN verifies that the licensee complies with its safety management system, especially in terms of organization and resources, and has established a cartography of the skills/competencies and the necessary resources as defined in its management system. The order enforces the licensee to implement himself the supervision of its contractors. The responsibility of the QA auditing of EDF

and/or its contractors is first of all provided by EDF.

Q.No	Country	Article	Ref. in National Report
113	Korea, Republic of	Article 13	4.1.2, page 86

Question/ Comment
It is mentioned in Clause 13.4.1.2 that "the role of ASN is to verify that EDF assumes its responsibility for the safety of its facilities by implementing a quality approach and in particular by monitoring the condition in which this subcontracting takes place".
Does ASN impose the limit of subcontracting level on EDF? If yes, what is the regulatory requirements for that?

Answer
ASN doesn't impose legally the limit of subcontracting level on the EDF. Meanwhile, EDF decided to limit itself the subcontracting level to three.

Q.No	Country	Article	Ref. in National Report
114	Switzerland	Article 13	13.4.1.2

Question/ Comment
The report states the quality aspects related to the use of contractors.

The issue of monitoring the condition in which subcontracting takes place is discussed in the here mentioned chapter. However, the FSOH analysis within the framework of the stress tests did indicate that the organisation of subcontracting is a major and difficult issue (see chapter 12, pape 82). How does EDF fulfill its responsibility to monitoring the entire supply chain? I.e. how does EDF monitoring an entire supply chain and what are the methods applied and difficulties encountered ?

Answer
When a contractor is working with EDF, an assessment is realized to evaluate his capability to fulfil all the expectations - When the contractor is working in the plant, he is also evaluated on how he fulfils these expectations in working - in case of difficulties, actions plan is applied and controlled. In any case, a yearly feed back is made.

Q.No	Country	Article	Ref. in National Report
115	Turkey	Article 13	83

Question/ Comment
As one of the major nuclear industry country, France should have a strong safety culture concept in its all nuclear facilities and activities according to IAEA GS-R-3 standard. Could you give information about how France establishing safety culture in its all nuclear facilities and activities?

Answer
For more than 10 years, EDF has developed a model of excellence based on European Foundation For Quality Management (EFQM) to increase at the same time the targets to continuously develop safety culture and the way of reaching them including for example leadership, involvement persons, facts management. This model has been developed as a best performance practice with tools and specifics structures. Every year, each NPP makes a self evaluation to measure how they implement EFQM model with safety as the first priority. An integrated policy is implemented with safety as the overriding priority and a specific management for safety guide with a specific questionnaire at each management level. It has been defined how NPPs must do their annual safety diagnosis (on

	management, results and specific topics in link with experience feed back, defence in depth....) and these analyses have been included also in a national review safety process. NPPs realise annually their own safety analysis review and a global strategic review. As a conclusion, EDF has implemented GSR3 into its own requirements.		
Q.No 116	Country United Arab Emirates	Article Article 13	Ref. in National Report 84
Question/ Comment	The National Report of France mentions, “The stipulation of a subcontracting system with a maximum of three tiers, including the contract holder, must apply to any contract, in compliance with EDF’s undertakings for new calls for bids and for currently ongoing contracts. The social specifications were examined by the performance-working group of the nuclear sector strategic committee set up by the Government in January 2012. The social specifications comprises rules common to all players in the sector, designed to guarantee know-how, skills, training and qualification, as well as the adoption of nuclear safety, radiation protection, occupational risk prevention and quality of life at work as fundamental, essential criteria. These social specifications and the proposals for regulatory changes were sent to the Prime Minister, the Minister for Industrial Renewal and the Minister for Ecology on 20th July 2012. It was also presented to the Steering Committee for Social Organisational and Human Factors, chaired by ASN, on the 9th November 2012. Implementation of these social specifications in the contracts will begin in financial year 2013.” Could France elaborate on how the subcontracting system will be enforced and further explain the meaning of the “social specifications”, and its likely impact on nuclear safety?		
Answer	The social specifications are now part of the expectations required for each contractor working with EDF. The initial assessment to be able to work for EDF and the renewal assessment to maintain this certification takes into account the social specifications. Part of the contractors expectations is his capacity to drive and control his subcontractors according to the EDF expectations. When working in the plant, the contractors management of his subcontractors is also assessed. The social specifications give, to all the contractors and subcontractors, expectations and responsibilities contributing to safety culture which makes a big step with other part of the industry . The social specifications are expectations common to all actors in the field, designed to guarantee know-how, skills, training and qualification, as well as the adoption of nuclear safety, radiation protection, occupational risk prevention and quality of life at work as fundamental and essential criteria. The implementation of the social specifications is controlled with specific assessments, at the initial assessment of the contractors to get certification, at the renewal assessment of the contractors to maintain certification, and each time when working on the plant. As the social specifications are relatively new and in progress for implementation, a specific steering committee is in place to follow this initial phase.		
Q.No 117	Country United Arab Emirates	Article Article 13	Ref. in National Report 86
Question/ Comment	Please describe the regulatory controls and processes that ASN has in place to address the potential supply of counterfeit or fraudulent material and components in particular for new NPP construction.		

Answer	ASN performs a specific and in depth control of the construction of the material which are used in the primary and in the secondary systems. This control is implemented through dedicated inspections in the manufacturers' workshops as well as their subcontractors. In 2013, ASN carried out 15 inspections of manufacturers and their subcontractors, and 15 inspections for the surveillance of third party bodies. Third party bodies carried out in 2013 around 2700 inspections related to the conception and the manufacturing of equipment for the Flamanville 3 EPR reactor and around 2000 inspections related to the conception and the manufacturing of steam generators for the in service reactors. For the other kinds of equipment, ASN controls the safety management system of the licensee which has to show that he can prevent, detect and eliminate the use of counterfeit items in NPPs. For items used in the safety of the NPP, the licensee safety management system is based upon the following pieces of information : - the spare procurement for all NPP is centralized; the department in charge of this activity writes technical specifications, identifies related risks, checks the spare in the factories and in the NPPs, monitor the activities in the NPPs. - each spare is identified in a so-called "reference file" (which contains the technical draws, quality controls ...), held by the manufacturer, and is referenced in the database of the licensee with its aim related to the safety. The most sensitive spares can only be supplied by few and accredited manufacturers - when a manufacturer wants to modify its spare, it has to change the "reference file" to determine to what extend it is likely to jeopardize the nuclear safety. - The licensee inspects the manufacturer and its contractors as necessary.

Q.No	Country	Article	Ref. in National Report
118	Belgium	Article 14.1	chap. E, sec. 14.3.1.2, page 102

Question/ Comment	- In § 14.3.1.2 of the French National Report, it is mentioned that ASN will request that the licensees of research reactors produce probabilistic safety assessments. Further, in § 14.3.3.1 it is indicated that the specificity of each research reactor makes it hard to carry out PSAs, which are therefore not performed for these facilities, but that ASN will request probabilistic safety analyses. In § 14.3.3.2, it is further mentioned that for the ILL high-flux reactor the Licensee proposes to use PSA. Our question relates as well to the present situation, as to plans for the Future. Concerning the present situation: • Are there at present no PSAs performed for any of the research reactors (which seems to be suggested in § 14.3.3.1)? • To which extent is this (or not) foreseen by the Periodic Safety Reviews of the research reactors? • If there are any PSA for research reactors, what is there scope?

	Concerning the future situation: • For the JHR, now under construction, what has been the role of PSA in granting the construction license? • What will be the role of PSA in granting the license for operation of the JHR?
Answer	Current situation: Currently , the demonstration of nuclear safety research reactors has some probabilistic analyses of accidents, but they are specific and different probabilistic safety assessment (PSA) as performed for nuclear power reactors. Indeed, such studies are difficult regarding the specificity of each research reactor. Probabilistic analysis for research reactors is rather based on a specific approach and proportional to the magnitude of the risks inherent in installation. The order of 7 February 2012 setting the general rules relative to BNI requires that from 1st July 2015 , the safety demonstration includes probabilistic analyzes of accidents and their consequences, unless the operator demonstrates that this is irrelevant (safety reviews , requests for authorization to create , change...). Prospect: The safety demonstration of the JHR validated for the issue of the decree authorizing creation includes occasional probabilistic analyzes accidents. Commissioning of the reactor is based on an evaluation of the compliance of the installation with the requirements set at the time of the decree authorizing creation. It will not be reassessed for commissioning. The systematic integration of probabilistic analyzes of accidents and their consequences in the safety demonstration, unless the operator demonstrates that this is not relevant will not be requested. However this should be stated within the next safety review.

Q.No	Country	Article	Ref. in National Report
119	Belgium	Article 14.1	chap. E, sec. 14.3.2, page 102

Question/ Comment	- In § 14.3.2 it is mentioned that the PSA plays a role in the Cost-Benefit-Safety approach used during a review. Is this an approach developed and applied by the Licensee of the NPP or by ASN? What are the basic principles of the Cost-Benefit-Safety approach?
Answer	The Cost- Safety Benefit approach consists on the one hand in the evaluation of the safety benefits of a modification based on its safety impacts on : -core damage frequency, -large early radioactive releases frequency, -late releases frequencies with and without venting after a severe accident, -basemat breakthrough frequency; -radiological consequences of accident without core melt. On the other hand the total cost of the modification (definition and studies, implementation, staff training, impact on operation & maintenance costs and on the plant availability) is assessed. During Periodic Safety Review, the modifications are ranked according to their safety benefit versus cost ratio. The last ranked modifications are of poor safety interest. They increase the global cost of the PSR without really improving the safety level of the

plant. If there is no other specific reasons to keep them, they can be rejected.
The Cost-Safety Benefit process is performed by the licensee, and the results submitted to the ASN licensing process.

Q.No	Country	Article	Ref. in National Report
120	Belgium	Article 14.1	chap. E, 14.3.1.2 page 102

Question/ Comment -
Article 3.3 of Part I of the BNI order, specify that the nuclear safety case for research reactors shall also include probabilistic assessments of accidents and their consequences, unless the licensee demonstrates that this is irrelevant. In 14.3.3.1 page 103 you mention "With regard to research reactors, the specificity of each one makes it hard to carry out PSAs, which are not therefore performed for these facilities, but ASN will request probabilistic safety analyses. Is this action already planned? What is the status of the PSA for JHR ?

Answer The safety demonstration JHR validated at the time of issuance of the decree authorizing creation includes occasional probabilistic analyzes accidents. Commissioning of the reactor is based on an evaluation of the compliance of the installation with the decree authorizing creation. It will not be a reassessment of safety. The systematic integration of probabilistic analyses of accidents and their consequences in the safety demonstration, unless the operator demonstrates that this is not relevant, will be stated within the next safety review.

Q.No	Country	Article	Ref. in National Report
121	Canada	Article 14.1	Page 101, Section 14.3.4

Question/ Comment ASN noted that "the reliability data for some equipment seem to have been produced from old experience feedback, even though the PSA must take account of changes to system characteristics, such as the equipment reliability data" and considers "that PSAs must use reliability data that are representative of the most recent operating experience feedback." Can ASN clarify whether measures have been undertaken to ensure updated PSAs will be consistent with their stated expectations?

Answer ASN asked EDF in a letter dated March 4, 2013 to survey permanently whether the reliability data used in the PSAs are representative when compared with the experience feedback and to update them if they are found to be insufficiently representative. EDF has taken an action to check that taking account the most recent operating feedback would not modify the results of the PSA performed for its 1300 MWe class reactors.

Q.No	Country	Article	Ref. in National Report
122	China	Article 14.1	section 14.2.1.4

Question/ Comment Description In section 14.2.1.4 ¡°On the occasion of the third ten-yearly outage of each reactor, this analysis leads to the production of a file clearing the reactor for continued operation.¡±
Question:

	What¡¯s the detail content of this file? What wouldbe analyzed and checked for operational units according to this file?
Answer	The file clearing the reactor for continued operation (DAPE) is sent by EDF one year before the third 10-year outage of each reactor. It is updated 6 months after the end of this outage: it includes the results of the controls performed during the outage and the status of the modifications associated with ageing management (modifications implemented during the outage or planned later with justification). In France, ageing management is mainly reviewed in the context of PSR and is based on a “double process”: a generic process associated with a reactor-by-reactor analysis. At the generic stage, EDF evaluates for each couple [degradation mechanism/component] several issues such as: maintenance and surveillance efficiency, reparation and replacement difficulties, potential or occurred mechanism. If this evaluation is not conclusive enough for at least one couple, EDF demonstrates in a specific file the capability of the concerned component to operate safely between the 3rd and the 4th 10-year outage. In the DAPE of the reactor, the licensee checks how the generic analysis is applicable to its unit and if there is any unit specificity which has been incompletely or not addressed in the generic analysis. If there is such specificity, then the licensee has to justify how related ageing is managed. Finally, the DAPE concludes about the continued operation of the reactor for the 10 years following the 3rd ten-year outage, including an ageing management program. This ageing program describes which actions are planned on the reactor in order to address all the unresolved ageing issues during the 10-year outage.

Q.No	Country	Article	Ref. in National Report
123	China	Article 14.1	section 14.2.1.6

Question/ Comment	Description in section 14.2.1.6"the licensee must notably identify situations leading to a sudden deterioration of the accident sequences (¡°cliff-edge effect¡±) and present measures for preventing them¡± Question: How to identify the situations leading to a sudden deterioration of the accident sequences? Please list the ¡°cliff-edge effect¡± events have been identified. What countermeasures have been done?
Answer	A cliff-edge effect corresponds to a large radioactive release, with long-lasting consequences on the environment. The situations, potentially leading to cliff-edge effects, which have been analysed in the stress-tests are the following : - Natural phenomena : earthquake, flooding, other phenomena related to flooding (storms, heavy rainfalls...) - Induced losses of safety systems on all the units of the plant : loss of ultimate heat sink (UHS), loss of all electrical power supplies, and combination of both. - Severe accidents One of the major conclusions of the stress-tests is that the existing protective measures against external natural phenomena provide sufficient margins to accommodate beyond design hazards. This result is the consequence of the Periodic Safety reassessments, performed in France since the early 1990's (1988 for CP0 units Fessenheim and Bugey) In addition, the stress-tests have confirmed the robustness of the existing means to cope with the combination of a loss of UHS and a

loss of electrical power supplies.
Nevertheless, in order to improve the autonomy of the plant in such extreme conditions, it has been decided to implement, as part of new ""hardened safety core"" SSC:
-an additional water source and make-up systems, to allow feeding of the steam generators, RCS injection and spent fuel pools water make-up;
-a new ultimate emergency diesel.

Q.No	Country	Article	Ref. in National Report
124	Korea, Republic of	Article 14.1	89

Question/ Comment: It is mentioned in page 89 that " ECS 1: Defining the structures and components of the 'hardened safety core'¡¦". Would you explain what are the typical structures, systems, and components which belong to "hardened safety core" ?

Answer: The hardened safety core relies on the implantation of additional SSC’s or existing SSC’s which are designed or checked against beyond design conditions (external hazards and a plant situation after this external hazard, with consideration of induced effects).
The global function of the hardened safety core is to guarantee ultimately basic safety function with reinforced means (criticality control, residual power evacuation, radiological confinement).
This hardened safety core relies on additional means. For reactors, this additional means comprises mainly:
• Bunkered diesel generator
• New ultimate heat sink
• New steam generator water feeding system
• Reinforced I&C for the steam generator and steam released valves
• Additional primary water feeding circuit
• Containment sump heat exchanger and related out-containment cooling system.
• Related I&C
• Reinforced primary pump seal protection system
• Containment isolation system…
These SSC need the operation of existing systems such as hydrogen recombiners that are in place on French plants for years.
For spent fuel pools, mainly:
• Bunkered diesel generator (same as reactors)
• New ultimate heat sink (same as reactors)
• Related I&C
• Reinforced Water feeding circuits
In addition, as part of the hardened safety core, an additional on site emergency response centre will be implemented to cope with

multi units accidental situations.
The implementation of the hardened safety core requires that existing SSC that have safety functions under specific conditions are checked regarding these conditions (reactor containment, PARs…), spent fuel pool structural integrity (under extreme hazard and induced effects such as heavy load drop).
This hardened safety core that relies on fixed means is also designed to be compatible and to house plugging systems to be supported if necessary by mobile means provided by some national repository.
On January 2014 ASN issued new resolutions to EDF related to the design and the implementation of the hardened safety core. Once translated in English, these decisions will be available on ASN web site: http://www.asn.fr.
The implementation of the most significant measures related to the hardened safety core (typically Bunkered diesel generator, New ultimate heat sink, additional on site emergency response centre) is forecasted by 2020 for the latest sites.

Q.No	Country	Article	Ref. in National Report
125	Romania	Article 14.1	14.1.1.1

Question/ Comment: What guidelines and criteria does the ASN use for the review of deterministic accident analysis, for design basis accidents and for design extension conditions? A reference where detailed information is already available would be useful.

Answer: Reviewing deterministic accident analyses is performed by specialists of our TSO (IRSN), based on the rules and criteria mentioned in the safety analysis report and previously accepted. The review of new rules, analysis methods or technical criteria is performed by IRSN and our standing expert committees according to the state of knowledge and international practices. For some topics, technical guides are available for the licensee and the TSO : the older are called "Basic Safety Rules" ; the more recent simply "ASN guides". A similar document called "Technical guidelines for the design and construction of the next generation of nuclear power plants with pressurized water reactors" has been specifically written for the next reactors, like EPR, It gives directions for the design and safety analysis of PWRs. Other guides have also been prepared concerning various safety related subjects, like probabilistic safety studies, protection against flood, pressure equipment, fire regulation, BNI decommissioning and dismantling. They can be obtained on request to ASN.

Q.No	Country	Article	Ref. in National Report
126	Switzerland	Article 14.1	P, 101

Question/ Comment: The safety case for these facilities is primarily based on a deterministic approach, whereby the licensee guarantees the facility's ability to withstand reference accidents. This approach is supplemented by probabilistic safety assessments (PSA)

In table 5 on page 102, the different generations of reactors has been grouped in relation to the PSA. Since the PSA is being used a cost-to-benefit decision-making tool, is it possible to indicate to what extent the PSA has enabled progress to be made in the CDF for each of the reactor generations included in Table 5?

Answer The PSA developments have allowed through the past 30 years to further improve French NPPs safety level by a progressive extension of their scope and field of applications.

The first PSA studies were performed in France in the early 90's on accidental sequences for internal initiating events, taking into account safety systems reliability and human factor. The back fit of these studies were on loss of power and sink situations as well as ATWS. Based on these models, PSA have been extended and upgraded by taking into account :

o More internal initiating events such as Common Cause Failures on 6,6 kV emergency supply buses;
o Shutdown states
o Further modelling of support systems such as ventilation systems

New fields for PSA have also been undertaken :

o Spent fuel events (loss of cooling, connected pipes failures) during the mid 2000 for the third 900 MWe safety reassessment and now for the third 1300 MWe safety reassessment;
o Level 2 PSA for core damage consequences and radiological releases;
o Internal hazards (internal flooding and fire) developed recently for the third 1300 MWe safety reassessment. These applications have confirmed the robustness of the French NPPs in terms of redundancy and reliability of the main systems involved in the safety demonstration.
o A first seismic PSA on a 1300 MWe NPP as part of the third safety reassessment on this type of reactor

This working program will continue in the frame of the French NPP life extension program mainly on aggressions and site long term situations affecting all the units.

For NPPs under operation, three main PSA applications are used to improve the safety and therefore decrease the CDF:

- The cost to safety benefits analysis that allows underlining the efficiency of modification to decrease the CDF. For example, an automatic activation signal of water supply into the primary circuit in shutdown states, which had been identified as an interesting modification by the cost-to-safety benefit decision-making tool, has been implemented.
- The analysis of dominant contributions to CDF. The identification of the main contributions to the core damage frequency highlights areas for which design and operation changes can be studied, and classified in order to target the priority work. For example, this work has led to diversification of the reactor scram function
- And finally the use of PSA for Beyond Design Basis situations. This analysis provides a renewed list of BDB, and defines the representative scenarios to be studied for each of the situations retained. For each scenario a specific Beyond Design Basis feature is identified using the PSA. This feature is considered as the most efficient feature to mitigate the corresponding accidental scenario. Consequently, these features benefit from specific operating requirement to ensure their availability. These analyses are updated at each periodic safety review.

	For NPP under construction, the PSA is mainly used to improve the design before the erection time. This analysis is mainly based on the analysis of dominant contribution to CDF. For example PSA highlights the necessity to add to the initial design a specific device in order to re-supply the EFWS tank.

Q.No	Country	Article	Ref. in National Report
127	Ukraine	Article 14.1	Table 5, page 102

Question/ Comment	Section 9.4 of the National Report of France on the "stress test" results provides information on completed and planned to perform PSA studies for different NPP designs: at the end of 2011 PSA for different NPP designs were fulfilled in a limited scope. In the table 5 of this report updated information related status of PSA studies is presented. It should be noted some progress, for instance, a seismic PSA for NPP Tricastin (900 MWe reactors) has been developed. Is it planned to develop additional PSA studies for different NPP designs, date of performance which is not related directly to the next PSR? Does full-scope PSA will be developed for all NPP designs in frame of next PSR (PSA level-1&2 for all internal and external events and for all operational modes, including seismic PSA? Additionally it should be noted that, in accordance with the ENSREG recommendations («Compilation of recommendations and suggestions ») periodically (especially in frame of the PSR) all external hazards shall be reassessed using the «state-of-the-art data and methods», probabilistic methods, including probabilistic safety assessment (PSA), are useful to supplement the deterministic methods.
Answer	For the next Periodic Safety Reviews (PSR) following the ones considered in Table 5 (i.e. 4th PSR of the 900 MWe NPPs, 4th PSR of the 1300 MWe NPPs, 3rd PSR of the N4 NPPs, 1rst PSR of EPR), different external events (including earthquake) will be considered and modelled from a probabilistic perspective when relevant, and considering: level L1&L2 PSA, all reactor operation modes and the spent fuel pool safety assessment. For each PSR, the relevance of the scope of the probabilistic safety analyses will be justified. Probabilistic safety assessments will be build on international state of the art approaches and will be adapted to the safety significance of the related risks. Note: some information presented in table 5 of the report has to be updated: - for the 900 Mwe NPPs a SMA -Seismic Margin Assessment was performed instead of a PSA -for the 3rd PSR of the 1300 MWe NPPS, all the events are considered -for the EPR commissioning application, level 2 assessment will only be limited to internal events (earthquake will not be considered).

Q.No	Country	Article	Ref. in National Report
128	Ukraine	Article 14.1	page 102

Question/	«In addition, the PSAs also contribute to two activities carried out during a review:

Comment	- definition of the complementary field; - the Cost – Benefit – Safety approach». Is it possible to provide more detailed information related to the «Cost-Benefit-Safety approach»? Does the Regulatory body support the Operator's decision on safety enhancement based on «Cost-Benefit-Safety approach»?
Answer	During a PSR, the operator has to define and propose safety enhancements, then this list of improvements is assessed by the regulatory body to determine if the safety objectives are achieved or not. To establish this list, the operator is free to use a cost-benefit based approach. EDF developed and submitted to ASN a Cost-Benefit-Safety approach, mainly based on PSAs, to be used during PSR. ASN considers this approach not mature enough to be yet taken into account : In particular, ASN considers that : - level 2 PSAs and PSHA need to be further developed and when ready integrated in the methodology - the methodology should not only focus on radiological release (health impact) but also take into account the economic impact of a nuclear accident (both with and without core melt).

Q.No	Country	Article	Ref. in National Report
129	Ukraine	Article 14.1	page 102-103

Question/ Comment	As mentioned in the report the Regulator requires the fulfillment of PSA for research reactors. It is emphasized that in the next PSR of the RHF (2017) a methodology for the safety assessment based on using both deterministic and probabilistic method has been proposed by Laue-Langevin Institute. Is it possible to provide general information concerning to this methodology taking into account specific issues for research reactors? Is it possible to provide general information concerning the approach for research reactors PSA (on which requirements of national documents, IAEA standards, documents from other countries related to PSA performing this approach is based)?
Answer	French requirements: According to the order of 7 February 2012 setting the general rules relative to BNI, the nuclear safety demonstration shall include probabilistic analyses of accidents and their consequences, unless the licensee demonstrates that this is irrelevant. They integrate the technical, organisational and human dimensions. Their application is based on an approach that is specific and proportional to the extent of the risks inherent to the installation. Unless otherwise specified by the regulator, these probabilistic analyses can be carried out in accordance with methods applied to the no nuclear industrial installations mentioned in article L. 512-1 of the environment code. The Basic Safety Rule number 2002-1 of 26 December 2002 supervises the development and the use of the probabilistic safety assessments (PSA) but it applies only for nuclear power reactors. The PSA carried out for nuclear power reactors are poorly suitable to research reactors given the specificity of each one. AEA standards: The § 3.29 of the IAEA safety standards guide No. SSG-20 is about probabilistic techniques which could be used to

supplement the deterministic safety assessments for research reactors. Probabilistic methodologies use the assumption that all accidents are possible and that any number of simultaneous failures may occur, although the probabilities may be very low. Some postulated accidents or combinations of accidents may have less dramatic consequences than the postulated accidents used in the deterministic methodology. However, when they are weighted by their likelihood, they may represent a significant risk and may impose different demands on the design. In addition, the deterministic approach has difficulties in effectively treating system interdependences (e.g. common cause failure), which probabilistic methods can address analytically and quantitatively. Application of probabilistic techniques also leads to significant improvements in the understanding of system behaviour and interactions, and of the role of operating personnel under accident conditions. These techniques may be indicated for some specific cases, which could be discussed between the operating organization and the regulatory body.

Methodology proposed to RHF by the licensee: Concerning the safety report of the RHF, the last methodology proposed by the licensee has to be modified to be agreed and authorized to be implemented by the regulator. It is based on a deterministic approach completed by probabilistic analyses which define postulated initiating events according to their frequency of occurrence. These probabilistic analyses are different to the probabilistic safety assessments (PSA) carried out for nuclear power reactors.

Q.No	Country	Article	Ref. in National Report
130	Ukraine	Article 14.1	page 88

Question/ Comment: Are there national general regulations/requirements for using validated and verified computer programs/codes for justification of the nuclear installations safety and detailed guidance for the validation and verification procedure? (The relevant requirements are determined by IAEA document GSR Part 4 "General Safety Requirements Part 4. Safety Assessment for Facilities and Activities", as well as other guidelines IAEA).
If so, what are the main Regulatory body objectives in the process on codes validation and verification?

Answer: The French order of February 7th 2012 laying down general rules for basic nuclear installations states that the demonstration of nuclear safety shall rely on the use of computational and modelling tools that are qualified for the domains in which they are used.
A working group, consisting of representatives of IRSN, EDF and ASN was launched in 2007 by ASN in order to prepare a preliminary set of guidelines about computing software qualification, This led in 2010 to the development of a compendium of best practices that could be used to ensure the qualification of scientific computing software. The field of study of this working group concerned the deterministic nuclear safety studies for PWRs.
ASN has decided in 2013 to create a wider working group for preparing a guide on the qualification of scientific computing software used in the demonstration of nuclear safety. The group will be formed soon.

Q.No	Country	Article	Ref. in National Report
131	United Kingdom	Article 14.1	Page 91, Section 14.2.1.3

Question/ Comment	The report states that the periodic safety review is an opportunity to review the condition of the facilities against applicable baseline safety requirements, e.g. is the degree of plant ageing acceptable for further operation. A second objective is to compare the requirements against modern standards, i.e. compare the requirements applicable to the NPP under review against those that would apply to a brand new NPP. Please clarify whether the periodic safety review also addresses the NPP's safety analysis to confirm that it remains fully appropriate and applicable in the light of the knowledge gained since the safety analysis was prepared? For example, such a safety analysis review would reveal whether fault frequency assumptions made many years ago remained valid or not given the many years of real operating plant experience now available. It might also reveal that some of the analysis techniques employed at the design stage have come into question and hence that some of the analysis ought to be repeated using better modern techniques.
Answer	Commissioning of the reactor is based on an evaluation of the compliance of the installation with the authorizing creation decree. It will not be a reassessment of safety. The systematic integration of probabilistic analyses of accidents and their consequences in the safety demonstration, unless the operator demonstrates that this is not relevant, will be stated within the next safety review.

Q.No	Country	Article	Ref. in National Report
132	Bulgaria	Article 14.2	p. 88-95

Question/ Comment	The report explains that the French Nuclear Regulator (ASN) requires the licensee to establish an integrated safety management system which to be able to maintain and continually improve safety during the operation of nuclear facilities. Bulgaria would appreciate if France provide some additional information in this respect, namely: - Which regulatory or licensing documents specify these requirements? - How does ASN monitor and verify the implementation of these requirements?
Answer	The order of February 7, 2012 requires the licensee to develop an integrated management system. This management system specifies measures implemented in terms of organization and resources. It also includes measures to identify safety important elements and safety important activities with the defined requirements for this elements or activities, to identify and solve gaps and significant events, to collect and take advantage of experience feedback, to define performance efficiency indicators. The licensee shall also regularly review its integrated management system to assess its performance, identify the potential improvements and implement approved improvements. These provisions will be completed by an ASN resolution on safety policy and integrated management system that should be published by the end of the first half of 2014. The correct implementation of these provisions is checked by ASN through different means: inspections on site to check the compliance of the operation with the safety management system, assessment of each significant event notified by the licensee to verify the corrective and preventive measures, periodic assessment of the national and international experience feedback (generally on a three years basis), periodic safety review process.

Q.No	Country	Article	Ref. in National Report

133	China	Article 14.2	section 14.2.2

Question/ Comment: Description In section 14.2.2 ¡± Following each ten-yearly outage, the baseline safety requirements for each plant series change in order to take account of the safety improvements made.¡±
Question:
How to set up the new level of ¡°base safety requirements for each plant series¡±? Normally the latest safety rules are the best choice, but the old plant series may not be able to satisfy these new requests. Please explain the consideration of it.

Answer: In the frame of the periodic safety review of a plant series, the existing safety requirements are evaluated on the basis of the best available techniques, the progress in technical knowledge, and the national and international experience feedback (in the fields of both design and operation).
This may lead to improve the safety requirements to implement the latest safety rules and objectives on older plant series, but it is not a systematic process. Actually, the upgrading is decided on the basis of a cost / safety benefit approach in order to identify the ""reasonably practicable modifications"" (see IAEA NSR-2/2 § 4.47). The general objective is to decrease the level of risk generated by the plant as low as reasonably possible, if technically possible, and in acceptable economic conditions.
The overall Periodic Safety Review aim at guaranteeing that the safety level of all the plants (including the older ones) is as close as possible to the safety level of Generation 3 reactors.

Q.No	Country	Article	Ref. in National Report
134	Czech Republic	Article 14.2	Page 90

Question/ Comment: Is in-service inspection programme (or changes within this programme) approved by a regulatory body?

Answer: In service inspection program is the responsibility of the operator. Concerning pressure equipments, third parties are only involved for control operations performed for which they receive an agreement from ASN. For pressure equipments with N1 level which includes the main primary and secondary circuit of the pressurized water reactors, ASN review theses programs with its technical support IRSN but without involvement of a third party.
On a general basis, the programme of maintenance planned by EDF during the reactor outages (including in service inspection) is reviewed by ASN before the beginning of the outage. Before operation start-up, the licensee has to send to the ASN an assessment describing all maintenance activities done during the outage and the results obtained. On that basis, ASN grants or refuse the license for start up.

For nuclear pressurized equipments (regulated by the interministerial order of 10th November 1999 relative to the monitoring of operations of the main primary system and the main secondary system of pressurized water nuclear reactors and/or the ministerial order of 12th December 2005 relative to nuclear pressure equipment), in-service inspection programme is approved on an individual

	basis by ASN. For the main primary system and the main secondary system, rules on the subject are defined by the order of 10th November 1999. The standard maintenance programs are approved by ASN. The licensees are requested to inform the ASN for any maintenance activities on the main primary and secondary systems of PWRs. In addition to this, a licensee planning a significant maintenance or repair on an equipment of the main systems has to obtain an ASN agreement. What should be considered as significant has been defined by regulation; it includes for example steam generator replacement, pressurizer heater replacement, steam generator chemical cleaning, relief valve flange seal resurfacing, latch mechanism replacement, pressurizer high pressure cleaning. For nuclear pressure equipments other than main primary and secondary systems of PWRs, rules are defined by the order of 12th December 2005. No need of ASN notification is requested for maintenance activities by this order. Nevertheless, for those equipments, significant maintenance or repairs have to be endorsed by an notified body accredited by ASN. In this case, what should be considered as significant has been defined by a professional guide (that guide being reviewed by ASN).		

Q.No	Country	Article	Ref. in National Report
135	Ukraine	Article 14.2	para 14.2.1.6 page 93

Question/ Comment	Were SAMG revised based on the results of stress-tests? What is the currently accepted scope of SAMG? Is the equipment employed in the SAM qualified?
Answer	SAMG, that is specific for each nuclear power plant, is periodically updated to include two types of evolutions: new equipment or intellectual updates (e.g. evolution of the scope). The EDF post Fukushima industrial programme following the stress tests will lead to both new equipment implementation (pumps, diesel, I&C, ...) and intellectual updates (Spent Fuel Pools monitoring...) for the French nuclear fleet in the next years. SAMG will then be updated gradually with the implementation of these evolutions. SAMG covers both the open and closed states of the primary circuit. Furthermore, some actions to prevent fuel uncovery in the spent fuel pool will be added to SAMG. The SAMG allows the use of both severe accident qualified , and non qualified, equipments. However, it is checked that the only use of SA qualified material is sufficient to manage a severe accident situation. Non qualified equipments are only used to optimize the SA operating strategies. Moreover the crisis team is able to validate key instrumentation outputs to avoid any misunderstanding due to instrumentations deviation.

Q.No	Country	Article	Ref. in National Report

136	United Arab Emirates	Article 14.2	92

Question/ Comment: The National report states that there has been a clarification of the requirements concerning the implementation of changes by the licensees and their review by ASN. Please describe further the nature of the clarification and any criteria or considerations used to determine which changes licensees can make without ASN review and which changes require ASN review.

Answer: In application of the essential principle fixed in article L 593-6 of Environment Code, the operator of BNI, is responsible for the safety of the installation. Furthermore, in application of articles 26 or 31 of the "decree 2007-1557 of 2 November 2007 concerning basic nuclear installations and the supervision of the transport of radioactive materials with respect to nuclear safety", when the operator envisages a modification to the installation such as to affect the interests mentioned in article L. 593-1 of the environment code, it declares or asks for authorisation to the regulatory body, forwarding a file containing all useful justification data, in particular the necessary updates of the elements in the authorisation decree or installation commissioning files and, in the event of a modification to the on-site emergency plan. The operator may not implement its project before a new authorisation if the modification is significant (like a rise in its maximum capacity or an operation with criticity risk for examples) or in other case, expiry of a six-month period, barring express approval by the regulatory body, which can also extend this period if it considers that further review or issue of additional requirements is necessary.

In addition to this legal framework, ASN is preparing some complementary requirements to detail modification management process.

The BNI procedures decree sets that the license defines and implements an integrated management system that ensures that the requirements relative to protection of the interests mentioned in article L. 593-1 of the environment code are always taken into account in any change concerning the installation. The ASN requires that the significant changes are carried out using methods and means that in principle should satisfy the requirements defined for the activities and the elements important for protection concerned, and enable this to be checked retrospectively. For minor modifications (like for an experimental device), the ASN may exonerate the licensee from the notification procedure provided that the operator sets up a system of internal checks offering sufficient guarantees of quality, independence and transparency. In application of articles 27 of the "decree 2007-1557 of 2 November 2007, these operations should not significantly compromise the installation's safety case nor significantly increase the installation’s impact on the interests mentioned in article L. 593-1 of the environment code.

Q.No	Country	Article	Ref. in National Report
137	Belgium	Article 15	E,sec. 15.2.2.2.5, p108, § 15.4.2.1,p117

Question/ Comment: -

Under § 15.1.2.2.5 “BNI discharges”, the report mentions the “process of reduction of BNI discharges at the source to reduce their quantity”. The report adds that the effort was initiated by the Authorities and implemented by the licensees.

	Under § 15.4.2.1., 8th alinea, the report mentions that the ASN will continue the work with the licensees to optimise discharges. This work of discharge reduction at the source is important because, once produced, the radioactive effluents have to be stored and eventually treated before discharge into the environment. In particular, the treatment of liquid effluents generates radioactive solid waste which has to be packaged in a proper way and for which a final disposal solution has to be developed, if not yet available. Moreover, it appears that the Authority plays an essential role as promoter of this process. It would be interesting to receive more information on this process of reduction discharges at the source.
Answer	The integrated approach required by this new system also applies to changes in the facilities and to reassessments of the facilities' safety. For these reassessments, Article L593-18 of Environmental Code stipulates that "the operator of a nuclear installation must periodically undertake a reassessment of the safety of its installation, in light of the best international practices". In addition, the Environmental Code provides that safety reassessments take place every ten years, subject to an exemption provided in the authorizing decree and justified by the particular features of the installation. Implementation of the new BNI system enables problems related to effluent discharges to be considered during safety reassessments. For existing facilities, ASN resolution 2013-DC-360 of 16th of July 2013 requires that registrants and licensees must periodically conduct a performance analysis of prevention and reduction of impacts caused by the nuclear facility in relation to the effectiveness of best available techniques including assessing performance differences. In case of discrepancy, registrants and licensees perform a techno-economic study to improve the performance obtained by the implementation of these best techniques. Furthermore, regulation states that the licensee shall favour reduction at source. When the best available techniques allow a significant reduction of the impacts they are implemented by the registrants and licensees if it’s technically and economically feasible.

Q.No	Country	Article	Ref. in National Report
138	India	Article 15	Page 112

Question/ Comment	It is reported that for the reduction of contamination in the primary system, controlled injection of Zinc is done. Can France provide information on the effectiveness of this method and resultant adverse consequences if any.
Answer	From a Radiation Protection point of view, the experimentation of zinc injection has not yet demonstrated any added value on dose reduction in France. Several causes are analysed, like the impact of load follow operating mode, delay in the injection at the BOC (beginning of cycle) for silica concentration concerns, or the age of the unit. An extended program is nevertheless implemented on 9 units for that purpose in order to get a broader panel and a better experience feedback.

Q.No	Country	Article	Ref. in National Report
139	Korea, Republic of	Article 15	1.3.1, 109

Question/ Comment	Are the dose limit and lifetime dose for part-time workers being managed? What method [EPD(electrical personal dosimeter) or TLD] is used to control and monitor exposure dose?

The place over 1 mSv/hr can be classified as a high radiation zone where a special access control is needed. Please explain the meaning of the sentence ¡°~ the hourly dose rate is liable to exceed 2 mSv¡±.

Answer: Concerning radiological protection, part-time workers are managed in the same way as full-time workers : all the requirements provided by the legislation and the regulations of the Labour code are to be applied to them, including the dose limits and the compliance of the dose received with these dose limits. Article R.4451-62 of the Labour Code requires that if the exposure is external, dosimetric monitoring of the workers is carried out by way of individual measurements known as passive dosimetery, using passive dosimeters. In addition to passive dosimetry, article R.4451-67 of the Labour Code requires that any worker called upon to perform an operation in a controlled area undergo an operational dosimetric monitoring, as a result of external exposure. Inside the controlled area, the employer delineates specially regulated or forbidden areas when the exposure is likely to exceed some of the levels set by a decision of ASN. A first specially regulated area is set up when the external exposure is likely to be more than 0,025 mSv/h. The employer takes all actions to ensure that specially regulated or forbidden areas are demarcated. These zones must have clear signage and are subject to specific access rules. The Labour Code prohibits the employment of temporary contract staff for the performance of work in areas where the dose rate is liable to exceed 2 mSv/h.

Q.No	Country	Article	Ref. in National Report
140	Pakistan	Article 15	Page 113

Question/ Comment: It is stated that the sites are classified according to “rating system”. Could France elaborate this “rating system?

Answer: In France, periodically, EDF Nuclear Inspectorate performs an Overall Excellence Assessment for each Nuclear Power Station. There is no global ranking at the end of the Overall Excellence Assessment (OEA). The ranking concerns only the themes of the 13 functional areas assessed during the OEA. For every functional area assessed we have 4 or 5 themes. For every theme, the ranking scale hold 7 level : Excellent, Good, Satisfactory, Average, Passable, Insufficient, Not acceptable.

Q.No	Country	Article	Ref. in National Report
141	Russian Federation	Article 15	Section 15.1

Question/ Comment: The Section says the European Commission has started the work to merge several EU directive in a single text, including those which deal with the basic radiation safety standards, protection of patients against medical exposures and control over high-activity sources. Currently, this proposal is under review at the Europe level and it is planned for publication in 2013.
Has this proposal been reviewed and published in 2013?

Answer: Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom (official journal of the European union L 13 / 17 January 2014).

Q.No	Country	Article	Ref. in National Report
142	Russian Federation	Article 15	Subsection 15.1.3.4

Question/ Comment: Is the information provided on doses, which are less than the regeneration level? How are they accounted for in determining collective doses?

Answer: Ministerial order dated 17 July 2013 concerning the individual medical file and the dosimetric surveillance of the workers exposed to ionising radiations provides requirements. For external dosimetry, the smallest measured effective dose and equivalent dose to the lens of the eye cannot be more than 0,10 mSv. For the extremities and the skin, it cannot be more than 0,50 mSv. Any value less than the recording limit of the dosimeter defined in the accreditation certificate is considered as zero and is transferred as zero by the approved dosimetry organisms to the national dose register (SISERI) . For internal dosimetry, the occupational physician provides the assessed internal effective or equivalent dose to the national dose register SISERI as far as he considers this dose as significant and, in any case, when this dose is equal or more than 1 mSv. Collective doses are then assessed taking into account all the doses saved in SISERI,

Q.No	Country	Article	Ref. in National Report
143	Switzerland	Article 15	15.1.2.4

Question/ Comment: One of the protective actions mentioned is the administration of stable iodine tablets, if the forecast dose to the throid is liable to exceed 50 mSv.

Is this administration valid for all age groups?

Answer: The intervention level for the administration of stable iodine is valid for all age groups. Nevertheless the posology depends on the age (2 tablets of 65mg for an adult (from the age of 12 and including pregnant women), 1 tablet for a child aged 3 to 12, 1/2 for a child aged 1 month to 3 year, 1/4 for new-born.

Q.No	Country	Article	Ref. in National Report
144	Switzerland	Article 15	15.1.3.1

Question/ Comment: The effective dose for workers is limit to 20 mSv. In some countries A and B workers are implemented dependent on the effective dose they will probably accumulate during the year.

Is the implementation of A and B workers foreseen in France?

Answer: In order to lay down the conditions under which radiological and medical surveillance are carried out, the workers likely to receive, in usual working conditions, an effective dose of more than 6 mSv per year or an equivalent dose of more than three tenths of the annual equivalent dose limits, are classified by the employer as category A, after consulting the occupational physician. Workers exposed to ionising radiation who are not in category A are classified as category B if they are subjected, within the context of their work, to exposure to ionising radiation likely to involve doses higher than one of the dose limits set for the members of the public.

Q.No	Country	Article	Ref. in National Report
145	Switzerland	Article 15	15.4.1.2

Question/ Comment: ASN considers that the average situation in the NPPs could be improved on a small number of points.

It would be helpful to mention these points in order to check whether it should be addressed as well in other countries.

Answer: The points that still need to be improved are the quality and integration of risk assessments, the contamination management in controlled areas, the monitoring of the application of radiation protection rules and the dissemination of lessons learned and best practices, so that they reach the workers.

Q.No	Country	Article	Ref. in National Report
146	Switzerland	Article 15	15.4.1.2

Question/ Comment: The collective dose is about 0,65 man*Sv per reactor .

The lowest and as well the highst collective doses accumulated shall be presented. Is there a correlation between collective dose and thermal power of a NPP?

Answer: The collective dose per unit in 2013 is between 0,14man*Sv per reactor to 1,5man*Sv per reactor, depending on the occurrence or kind of outage, and the amount of interventions planned during outage.
From EDF standpoint there is no relation between the thermal power of the NPP and the collective dose per unit. As for example, the average collective dose per unit for the 1300MW units fleet is lower than the average collective dose per unit for the 900 MW units fleet.

Q.No	Country	Article	Ref. in National Report
147	Ukraine	Article 15	para 15.4.2.1 page 117

Question/ Comment: What are the actual levels of individual doses of internal exposure of category A personnel at your NPPs (with the type of reactor in operation to be specified)?

Answer: Since 2008 no internal dose have been registered. This means that no internal exposure has resulted in more than a calculated dose superior of 0,5 mSv cumulated over 50 years. Very few traces of internal contamination are detected every year but calculated cumulated doses over 50 years are far lower than 0,5 mSv.

Q.No	Country	Article	Ref. in National Report
148	United Arab Emirates	Article 15	220

Question/ In Appendix 4 Figure 20: Synthesis of discharges from NPPs (1998 – 2011) shows a downward trend in the discharges of Iodes +

Comment	Fission and Activation products per unit of electricity generated, whereas there is a slight increase in Tritium and Carbon-14. As a fraction of total liquid discharges the H-3 and C-14 are now dominating in total MBq discharged. Please outline any considerations given to reduce the impact of C-14 and H-3, as the dominant isotopes released as reactor water quality and waste treatment technology improves?
Answer	ASN considers that the impact of releases are very low. ASN publishes in its annual report the dosimetric impact of all facilities (chapter IV: http://www.french-nuclear-safety.fr/index.php/English-version/ASN-s-publications/ASN-s-annual-reports). Despite this low impact, for existing facilities, ASN resolution 2013-DC-360 of 16th of July 2013 requires that registrants and licensees must periodically conduct a performance analysis of prevention and reduction of impacts caused by the nuclear facility in relation to the effectiveness of best available techniques including assessing performance differences. In case of discrepancy, registrants and licensees perform a techno-economic study to improve the performance obtained by the implementation of these best techniques. Furthermore, regulation states that the licensee shall favour reduction at source, When the best available techniques allow a significant reduction of the impacts, they are implemented by the registrants and licensees if it is technically and economically feasible. This process includes C-14 and H-3. For tritium, in 2008, ASN decided to set up two working groups to review current scientific knowledge about the environmental and health impact of tritium (“Tritium Impact” group), to review the sources and impact of tritium (“Tritium: Defence in Depth” group) and to issue recommendations, where necessary. The specific objectives of two groups were to explore the following topics: 1) the potential for tritium accumulation along the food chain and whether there is a need to reassess the health effects of beta radiation from tritium (“Impact” group), 2) consequences of the future increase in tritium discharges, and industrial solutions for tritium separation and sequestration from liquid or gaseous discharges (“Defence in Depth” group). ASN published this work in a White paper (http://livre-blanc-tritium.asn.fr). ASN publishes on this website, tritium releases of each facility, and the impact of tritium.

Q.No	Country	Article	Ref. in National Report
149	Canada	Article 16.1	Page 119, 16.1

Question/ Comment	The report states on page 119 that exercises are performed “regularly”. Can you elaborate on the frequency of exercises, such as annual, bi-annual, etc?
Answer	Each year around 10-12 exercises are planned in the annual calendar. In these exercises, both the site emergency plan and the off-site emergency plan are tested. The ASN, its TSO the IRSN, the prefect of the concerned area, the headquarters of the licensee and the plant are playing. In addition, the nuclear sites have to test their site emergency plan at least once a year. In addition, France participates to international exercises such as Convex, Ecurie or INEX.

Q.No	Country	Article	Ref. in National Report
150	China	Article 16.1	section 16.2.5.1.1

Question/	Description In section 16.2.5.1.1¡°It£¨IRSN£© must work in close collaboration with the licensee's technical teams in order to reach

Comment	converging views on analysis of the accident situation and to predict its development and consequences.¡± Question: If they cannot reach converging views, who and how to make decision?
Answer	IRSN and the licensee's technical teams must exchange to try to reach a common understanding of the situation and a converging analysis. If the assessments on the diagnosis and prognosis of the situation do not coincide ASN will take decisions based on the most realistic evaluations.

Q.No	Country	Article	Ref. in National Report
151	China	Article 16.1	section 16.3.1.1

Question/ Comment	Description In section16.3.1.1¡°the local emergency response team (ELC), more specifically in charge of the analyses of the state of the facility and predicting developments¡± Question: ELC is near the main control rooms of NPP, for example, Room L543 in LX Building of twin-unit NPP. What about multi-unit accident, as six-unit NPP in Gravelines, FRANCE? Is there another ELC available on site?
Answer	In case of a multi-unit event on a site with more than two units, the local emergency response team (ELC) will be located in the closest Crisis Technical Support Room (CTSR) of the first damaged unit, or, if relevant, in the CTSR of the unit in charge of site shared systems. Site CTSR will be located in new Emergency Control Centre to be build in the framework of Fukushima feedback.

Q.No	Country	Article	Ref. in National Report
152	Germany	Article 16.1	Page 134, Section 16.5.2

Question/ Comment	It is stated that the last preventive distribution campaign for iodine tablets dates back to 2009 and concerned the populations located within the zone covered by the PPIs (off-site emergency plans) around nuclear power plants. During this campaign, ASN distributed an information flyer on the monitoring of nuclear safety and radiation protection to 400,000 homes and 2,000 establishments open to the public. France describes in very detail the strategy for tablet distribution that aims to maximise the final coverage in the population. This strategy is recognized as good practice. What is the planned frequency for an update of such a distribution?
Answer	The iodine tablets are produced by the French Central Pharmacy of the Army with indication of the production date. The expiration date is not mentioned on the tablets boxes anymore. The initial conservation period was 48 months (4 years). There are currently studies under progress on the iodine stability. Their conclusions will enable to adapt and extend the expiration date in the future, probably up to 7 years. Since 1997, distribution campaigns took place in 2000, 2005 et 2009. The date of the next campaign is not yet defined.

Q.No	Country	Article	Ref. in National Report

153	Ireland	Article 16.1	Article 16.1.3.2, p 121

Question/ Comment	It is noted that the approach used in preparing plans relies primarily on a conservative theoretical approach leading to an estimation of the source terms, then calculating their dispersal in the environment, and finally assessing their radiological impact. Please provide further information on the bases used for estimating the source terms (e.g., what level of conservatism is used; what is the role of ASN/IRSN and the licensee in preparing determining the estimates).
Answer	In France, the IRSN, as TSO, calculates the source term based on the plant conditions and on the weather forecast. ASN is thus able to have an estimation independent from the one provided by the licensee. Then, the IRSN prepares a diagnosis, a prognosis and a "further failure prognosis" of the situation.

Q.No	Country	Article	Ref. in National Report
154	Ireland	Article 16.1	Article 16, p 118

Question/ Comment	With regards to the requirement to share information with neighbouring states that may be impacted by an emergency at one of France’s nuclear sites – what type of information has been shared with France’s neighbouring states?
Answer	ASN has bilateral conventions with several countries regarding the alert and information exchange. These conventions also mention the nature of information that should be shared both in the preparedness phase and in emergencies. These information can be related to the national emergency organisation, the network of measurements, contact details in routine and in emergencies, etc.

Q.No	Country	Article	Ref. in National Report
155	Japan	Article 16.1	p122

Question/ Comment	French report shows different action centers in an emergency, for example Fig.7, 8, 9. These centers locate remote places each other in France. How do you secure robust and reliable communication networks among those centers in emergency situation? Please show recent improvements and enhancements.
Answer	In emergencies, several actors activate their emergency centres. They are connected through different systems: analogical, numerical, satellite and dedicated interministerial networks. These networks also use different infrastructures. Should a network be out of order, another one would be used.

Q.No	Country	Article	Ref. in National Report
156	United Arab Emirates	Article 16.1	134

Question/ Comment	Please describe the criteria or references for defining the emergency planning zones referred to in section 16.3.5 including the a population protection zone (ZPP), heightened territorial surveillance zone (ZST) and evacuation zone. Does France consider Ingestion and Commodities Planning Distance (ICPD) for emergency planning?
Answer	Post-accident zoning is designed to provide a structuring framework within which actions to protect the population and manage

contamination across the territories affected by the accident can be instituted. The first post-accident zoning is established on the basis of a predictive model of future population exposure to the ambient radioactivity in the inhabited zones and contamination in the food chain, as a result of deposited radioactivity. This depends directly on the extent of the radioactive deposits, the persistence of which can vary substantially.

The public protection zone (ZPP) is defined as the area within which actions designed to reduce exposure to ambient radioactivity for residents of the said areas as low as reasonably achievable are warranted. This area is defined for the purpose of providing radiation protection for the population living in the most contaminated territories, based on dosimetric guidance values.

The initial definition of the ZPP will be made on the basis of assessment of projected doses likely to be received during the month following the end of release, without taking into account the effectiveness of the contamination reduction actions implemented in the area. The ZPP is in other words delineated based on the most disadvantageous of the two following exposure indicators:

- the projected effective dose received during the first month following the end of release, regardless of pathways of exposure, including ingestion of contaminated local foodstuffs, the guidance value used being approximately 10 mSv over the first month;
- the projected thyroid equivalent dose received over the course of the first month following the end of release, regardless of pathways of exposure, in particular ingestion of contaminated local foodstuffs, the selected guidance value being approximately 50 mSv over the first month.

The dosimetric guidance values are not to be interpreted as thresholds or limits. The uncertainties as to dose estimates are such that other factors than dose should be considered.

These other factors are connected with the conditions under which the actions envisioned are carried out in reality, and are best assessed at the local level.

Contextual factors may also make it appropriate to use more restrictive or higher values, or even to refrain from implementing any protection actions at all.

It may be that, across part of the ZPP, despite the ban on consumption of foodstuffs of local origin, exposure across the population may continue to be deemed too high, due to radioactivity deposited in the living environments. In this case, inhabitants must be displaced from the relevant part of the ZPP, probably for a longer duration, and a relocation perimeter (PE) must be established. The relocation perimeter shall be delineated based on the results of an assessment showing the projected effective doses over the first month following release, not taking into account the contaminated foodstuffs of local origin ingested, comparing them to a guidance value on the order of 10 mSv over the first month.

The heightened territorial surveillance zone (ZST) extends beyond the borders of the public protection zone. As the emergency phase comes to an end, the ZST is also delineated, using forecast
assessments derived from models of the transfers of radioactivity deposited in farming areas. It is characterised by lower environmental contamination that does not require the automatic
implementation of population protective actions. This contamination is nonetheless significant and can affect in particular foodstuffs and agricultural products, substantiating the institution of
specific systems to monitor the radiological quality of the relevant products. In some agricultural products and foodstuffs, contamination may exceed, albeit temporarily,
the maximum permitted levels (NMA), considered of regulatory value and set at the European level to regulate the placing on the market of the said foodstuffs
(Council Regulation (EURATOM) No. 3954/87 adopted on 28 December 1987 as modified by Regulation (EURATOM) n° 2218/89 adopted on 18 July 1989, which sets NMA
of contamination in foodstuffs following a nuclear accident).

For more details : www.asn.fr

Q.No	Country	Article	Ref. in National Report
157	United Arab Emirates	Article 16.1	135

Question/ Comment: The “post-accident” phase concerns the longer-term management of the consequences of a lasting contamination of the environment by radioactive substances following a nuclear accident. It concerns dealing with varied consequences (economic, health, social), in the short term or even the long term, with a view to restoring an acceptable situation. Is the significant psychological impact for off-site population considered as a one of the consequences of nuclear accidents? Please provide details about the measures for reducing psychological impacts on the affected population.

Answer: The creation of reception and information centres (CAIs) is one of the first actions on which the public authorities must decide as the emergency phase comes to an end. The CAIs must be the place of choice for providing personalised information to those involved, in particular on issues related to radiation protection, social support or compensation.

Epidemiological studies have emphasised the importance of the psychological impact of changes in living environment (evacuations, relocation, alteration to environment, suspension of agricultural activity, etc.) and the uncertainties that come along with them. This impact appears to depend little on the importance of contamination in the environment, but is connected more to the perception of radiological risk and other accident consequences, as well as the quality and speed of the responses provided, both from the healthcare and social standpoints.

In order to mitigate the negative aspects of these reactions, which are normal under such circumstances, clear information must be issued quickly, about the risks, recommendations on individual protection and places where support can be found (CAI). A medical-psychological emergency units (CUMP) (or, at least, a unit capable of directing the affected population to mental health professionals) provides listening services and psychological support. Recommendations should be issued to professionals working in the healthcare networks so that they can pay close attention to potential signs of stress overload or psychological exhaustion and, if necessary, guide people requiring additional care such as an internal contamination measurement that can reassure the most worried people.

Q.No	Country	Article	Ref. in National Report
158	China	Article 16.2	section16.5.2

Question/ Comment	Description In section16.5.2¡°The Ministry for Health thus ordered the manufacture of 110 million 65 mg tablets, which have been shipped to the regional platforms¡± Question: How long the tablets would be renewed?
Answer	The iodine tablets are produced by the French Central Pharmacy of the Army with indication of the production date. The expiration date is not mentioned on the tablets boxes anymore. The initial conservation period was of 48 months (4 years). There are current studies under progress on the iodine stability. Their conclusions will enable to adapt and extend the expiration date in the future most probably up to 7 years. Since 1997, campaigns of distribution took place in 2000, 2005 et 2009. The date of the next campaign has not been defined yet.

Q.No	Country	Article	Ref. in National Report
159	Romania	Article 16.2	16.2

Question/ Comment	Does the ASN have the responsibility to perform an independent estimation of the radiological consequences of an accident? If yes, what are the sources of information used, the methods and the tools employed to perform the necessary calculations? Is it the IRSN staff that performs the calculations?
Answer	In France, this is the IRSN, as TSO, which performs an independent estimation of the radiological consequences of an accident. IRSN is connected to the plants and receives information from the different sensors. The estimation is based mainly on the plant conditions and on the weather forecast.

Q.No	Country	Article	Ref. in National Report
160	Ireland	Article 16.3	Article 16.4, p.130

Question/	It is noted that France has a extensive programme of annual emergency exercises and that there are plans to include local populations

Comment	in the accident (16.4.3). What has been the experience to date in organising such exercises involving potentially affected populations – or when are they planned to begin?
Answer	Each year around 10-12 exercises are planned. The involvement of the population is rather rare. It depends on the annual objectives defined jointly by Prime minister service, Home office (ministère de l'intérieur), ASN and its counterpart for defence activities (DSND). It also depends on the prefecture organising the exercise. One of the main exercise which involved the population was organised in 2011 for the exercise of Gravelines. In particular, a real evacuation of the population (already informed of the exercise) was organized including schools. For 2014, one of the annual objectives is to test real civil protection actions.

Q.No	Country	Article	Ref. in National Report
161	Ireland	Article 16.3	Article 16.5.3, p 135

Question/ Comment	The post-accidental preparedness programme that has been carried out by France is very impressive and should be considered as a candidate for a good practice.
Answer	France really appreciates this comment.

Q.No	Country	Article	Ref. in National Report
162	Switzerland	Article 16.3	16.3.1.2, P, 128

Question/ Comment	According to the report, EDF is setting up the Nuclear Rapid Intervention Force (FARN). Is it planned to include the FARN in emergency exercises and if so, when?
Answer	The teams that compose the Nuclear Rapid Intervention Force (FARN) are trained by the operator. They also take the opportunity of national emergency exercises to train but independently from the technical scenario. For 2014, the emergency exercises with the participation of the FARN have not yet been identified.

Q.No	Country	Article	Ref. in National Report
163	Switzerland	Article 16.3	16.4.3/16.5..2, P, 133

Question/ Comment	According to the report, iodine tablets are distributed preventively to the population in a radius of 10 km around the NPPs. In total 110 million tablet have been distributed to regional platforms. For the rest of the population around a NPP how is the distribution organized? How is the distribution of iodine tablets organized for the area beyond the 10 km radius around a NPP and what is the assumed time needed for a distribution of these tablets to the population in case of an emergency?
Answer	Outside the radius of 10km where a pre-distribution is organized regularly, a circular signed in 2011 jointly by the Ministry of Health and the Ministry of Interior defines the organisation of the distribution of iodine tablets in emergencies. There are regional stockpiles

at several wholesalers premises. The prefects are requested by the circular to prepare specific plans mentioning the locations of the iodine tablets, the actors and methods of distribution in emergencies. In addition, there is a national stockpile at the French Central Pharmacy of the Army.

Q.No	Country	Article	Ref. in National Report
164	Canada	Article 17.1	Page 137

Question/ Comment: Please confirm whether the re-assessment of safety performed during the Periodic Safety Reviews results in a reconsideration of all of the safety-related characteristics, in particular external hazards, that were used in the approval of the site. If some characteristics are not re-confirmed, please explain the basis for these characteristics being exempt.

Answer: During the periodic safety review (PSR), different safety topics are identified for reassessment. This identification is mainly based on the objectives for safety improvements, on the operating experience from the reactor, from other French NPPs and from other reactors abroad, and on the most recent safety requirements for new reactors. For example, more than 20 safety topics were identified for the 3rd PSR of 900 MWe NPPs (e.g. severe accidents management, probabilistic safety analysis, reactor containment…). For those issues, all safety-related characteristics are reconsidered.
Regarding external hazards, they are reassessed during the PSR, or in the context of specific studies (for example, external flooding after the event in Blayais in 1999, extreme heat after the heat wave in 2003, lessons learnt from the Fukushima-Daiichi accident…).
Moreover, as required in the draft ASN resolution “Periodic safety review” that should be issued in the forthcoming months, all safety-related characteristics from the safety report have at least to be checked, and the need for not reconsidering them has to be justified.

Q.No	Country	Article	Ref. in National Report
165	Japan	Article 17.1	p141

Question/ Comment: French report says, gEDF plans providing a diesel generator set for each plant unit to supply the instrumentation & control and control room lighting.h
How are SSE and DBE for this diesel generator specified in order to accommodate to total loss of electrical power supplies? How are more intense SSE and DBE sufficient for this generator compared to the conventional plant unit emergency generators?

Answer: Before the implementation of the main Ultimate Diesel Generator, belonging to the Hardened Safety Core, EDF has already implemented a smaller and specific diesel generator in order to supply the instrumentation & control room lighting, and provide minimum electric sources in case of total loss of electrical power (SBO). This diesel was implemented before the end of June 2013 on the 58 French reactors. This diesel doesn’t belong to the Hardened Safety Core and it has no seismic requirement.

All the Hardened Safety Core SSC (Systems Structures and Components, including the new Ultimate Diesel Generators) have a specific Safe Shutdown Earthquake level called ""SND"". The ""SND"" is at least 1.5 times higher than the design SSE of the other safety systems of the plant and will cover the seismic events with a return period of 20 000 years.

As an additional conservatism the new SSC of the Hardened Safety Core (and the Diesel Generators are in this case) will be designed using a broad band design spectrum (EUR type). For a low seismicity site the new design will be similar to the existing EDGs. A design level of 0.2 g is more than sufficient.
For medium seismicity sites the new diesels may be robust to a seismic level of about 2.25 times higher than the existing ones : 2,25=1.5 (which is the difference between the SSE and the SND) x 1.5 (which is the margin between the SND and the broad band design spectrum).

Q.No	Country	Article	Ref. in National Report
166	Japan	Article 17.1	p143

Question/ Comment
French report says, □gin order to prevent any ingress of water, EDF has implemented a protected volume approach.□h
How higher water level is assumed than conventional design base level? Please show the basic idea of the assumed water level?

Answer
The volumetric protection (or ""protected volume"") approach involves making water-tight certain plant rooms which are situated below the platform ground-level and which contain safety related equipments. Protection is ensured by water-tightness of all openings and penetrations (pipe work, cables ...) on the external walls of these rooms.
Volumetric protection ensures that the rooms situated below ground-level remain water-tight in the following reference flooding events:
• Increase in groundwater level where, in general, maximum values are seen during the maximum flooding events considered in the safety case;
• Infiltration of water (internal or external origin) which is likely to come from non-classified rooms, which do not house functions whose protection against flooding, is required.
Protection is insured up to a height of 0 m (ground level) for buildings located on the nuclear island platform.

For most sites the flood level considered in the safety is lower than the level of the platform; the height of the volumetric protection therefore affords a safety margin with respects to the reference flood level. The safety margin can vary from tens of centimetres to several metres, according to the site.

On certain sites, the successive Periodic Safety reassessments undertaken since their initial design have resulted in the safety case reference flood level being revised upwards, above the platform level. In these situations, the protection of safety-classified plant rooms is ensured, either:
• By dykes or perimeter walls encircling the buildings requiring protection.
• By increasing the height of the volumetric protection over and above the reference flood level (a minimum margin of 20 cm is taken), the sealing of the different access doors being afforded by flood barriers put in place by the operator upon receipt of a flood warning.

Q.No	Country	Article	Ref. in National Report
167	Switzerland	Article 17.1	Section 17

Question/ Comment

Is there a design basis requirement for the French nuclear power plants regarding extreme temperatures?

Answer

Regarding extreme cold temperatures, the design basis steady-state temperature is -15°C and depends on the location of the plant for short time temperature (7 hours is considered). Regarding extreme hot temperature, there are no design basis temperature. Following 2003 and 2006 heat waves, EDF developed a framework taking into account global warming and heat waves depending on the location of the plant.

The art. 3.6 of the ministerial order of 7 February 2012 requires to take into account extreme climatic conditions.

Q.No	Country	Article	Ref. in National Report
168	Switzerland	Article 17.1	P, 144-146

Question/ Comment

Could you provide the exact reference of the "Snow and Wind Rules" mentioned on Pages 144 and 146 of the report?

Answer

The reference is "Règles définissant les effets de la neige et du vent sur les contructions et annexes" edited by CSTB - AFNOR standard DTU P 06-002

Q.No	Country	Article	Ref. in National Report
169	Switzerland	Article 17.1	P, 149

Question/ Comment

On Page 149, it is mentioned that ASN published a new guide on protection against external flood risk for nuclear facilities, which integrates the recommendations of RFS I.2.e and the experience feedback from the flooding of the Blayais site in 1999. Could you provide the reference to this document? Is this ASN guide publicly available?

Answer

The ASN's guide to help licensees or applicants take better account of the flood risk in nuclear facilities is available on the internet website of ASN : http://www.french-nuclear-safety.fr/index.php/English-version/News-releases/2013/ASN-Guide-relative-to-the-protection-against-external-flooding

Q.No	Country	Article	Ref. in National Report
170	Switzerland	Article 17.1	P, 143

Question/ Comment

How were the flooding heights at NPP sites determined? Were 1D or 2D simulations used? If 2D simulations were used, which software was applied?

Answer

According to the modelled river and the watershed configuration, EDF used 1D models or 2D models. For most of the sites, EDF used 1D models to simulate the flow to route the flood until the site (long linear of river) and 2D models to calculate the flood height at the site. For 2D simulations, EDF uses the Telemac software.

Q.No	Country	Article	Ref. in National Report
171	Ukraine	Article 17.1	page 148

Question/ Comment ASN extended for six months (until 31.12.2013) the requirement related to carry out a study to compare the seismic instrumentation currently used in France with that used internationally. A decision on necessity of seismic monitoring equipment replacement shall be made after the ASN review of this study (ASN letter to EDF further to the meeting of the advisory committee of experts on reactors in November 2011: CODEP-DCN-2012-020754 of 26 June 2012, French National Action Plan based on stress-tests results).
Are the results (with appropriate suggestions and the schedule of measures) already available for ASN?
If so, what are the main results of this study?

Answer The EDF fleet seismic instrumentation is designed and operated according to the ASN Fundamental Safety Rule ""RFS 1.3.b"".
The comparison, from official documents up to date, has been performed using two different ways :
- The first one is evaluation of IAEA recommendations based on international good practices,
- The second one is the evaluation of the practice of European countries close to France and with moderate seismicity like Germany or Switzerland.
The result of this study shows that the French practice is suitable compared to the other ones, so that EDF hasn't planned any evolution of its seismic instrumentation.
These studies have been sent to ASN at the end of December 2013 as requested, and are under ASN review.

Q.No	Country	Article	Ref. in National Report
172	Belgium	Article 17.2	chap. F, sec. 17.2.3, pages 144-146

Question/ Comment -
Under § 17.2.3, the report discusses extreme climatic conditions associated to wind, hail, lightning and snow, but not to rain, notably in relation to the assessment of the capacity of sewer systems. What is the status of this last concern in France and are there, or what are the related ASN demands?

Answer The rains was considered in the "flooding" part of the stress test. In the stress tests, EDF considered a doubled maximum high-intensity rainfall (PFI), and a combination of a PFI lasting 60 minutes with a complete blockage of the site's rainwater drainage network outlets.

Following the stress tests, ASN required:
Before 31 December 2013, the licensee shall present ASN with the modifications it intends to make to reinforce, before 31 December 2017, the protection of the facilities against the risk of flooding beyond the baseline requirement in effect on 1 January 2012, for example by raising the protection volume to protect against situations of total loss of the heat sink or electrical power supplies, for the beyond design-basis scenarios, such as:

	- maximum rainfall, - flooding resulting from failure of on-site equipment under the effects of an earthquake.

Q.No	Country	Article	Ref. in National Report
173	India	Article 17.2	Article 17.2.1 Page 140

Question/ Comment	It is stated – 'In response to an ASN requirement, EDF studied the advantages and drawbacks associated with the installation of an Automatic Reactor Trip System (AAR) in the event of a seismic stress. EDF concluded that the safety benefit resulting from an earthquake triggered AAR function justified its implementation on the reactor fleet in operation. The earthquake-triggered AAR is beneficial for emergency shutdown rod drop and clarifies the initial conditions of the plant units following an earthquake.' Will France please indicate technical reasons in favour of this decision?
Answer	The main benefit of an Automatic Reactor Trip in the event of a seismic stress is to anticipate the scram relatively to the already existing scram I&C signals (mainly by loss of off site power-LOOP). There are some drawbacks which have been considered minor in regard to the safety improvement of such a modification : o the plant automatic trip could destabilize the electric balance of the national network but in such a case the electric network itself may not be fully operational following a significant earthquake, o the scram of all the reactors of the site will induce simultaneous operations to bring the reactors in safe shutdown conditions, which may be challenging for the on site teams.

Q.No	Country	Article	Ref. in National Report
174	India	Article 17.2	17.2.3, Page 144

Question/ Comment	With reference to external natural events, have there been any studies conducted on (i) hazards associated with extreme low temperature causing freezing and subsequent loss of UHS and (ii) safety of nuclear installations against volcanic event?
Answer	i) EDF analyzes the following hazards associations : - frazil and extreme low temperature and LOOP (extreme low temperature could generate a LOOP -Loss Of Off-site Power-) - extreme low temperature causing freezing or frazil which induces low water level or low cooling water flow at the intake. Taking into account dedicated equipments and operating dispositions, EDF considers that extreme temperature is not likely to induce subsequent loss of UHS. Nevertheless, LUHS (Loss of Ultimate Heat Sink) is taken into account in the design and operating procedures of the fleet ii) Volcanic risk is not considered in the safety case of French nuclear installations because no plants are located in active volcanic areas.

Q.No	Country	Article	Ref. in National Report
175	Korea, Republic of	Article 17.2	138

Question/ Comment	Are there any NPP monitoring systems operated by either the utility or the regulatory body for site characteristics, such as earthquakes, surface faulting, groundwater elevation, slope failures, foundation subsidence etc., and are these information open to the public, if any?
Answer	1) Earthquakes In accordance with the Fundamental Safety Rule RFS I.3.b, a seismic monitoring equips each NPP site. This instrumentation is composed by : . Tri-axial accelerometers located at different places : two in the reactor building, one in another nuclear island building whose foundation is separated from the reactor building and one at free-field, . time histories recording system attached to the accelerometers, . alarm in every unit control room. If soil conditions are heterogeneous, another accelerometer is added at free-field and on the reactor building of each unit. This monitoring system helps the operator to decide to shutdown the NPP in case of earthquake to prevent damage on structure or equipment. A seismic trigger of 0,01 g starts all recordings and actuates the alarm in all control rooms. All the units of the NPP are shut down when the peak ground acceleration of half of the design spectrum adapted to the site is exceeded at any of the measuring points. The resumption of operations can be performed only after justification to the nuclear regulatory body (ASN) of the innocuousness of the earthquake on the future behaviour of the plant. 2) Foundation Subsidence A monitoring composed by topographic measurements is managed locally. 3) Groundwater elevation Each NPP site is equipped by a network of piezometers for both purposes of groundwater level measurement and environmental monitoring. 4) Others site characteristics such as surface faulting, slope failures Our NPPs do not require any monitoring system for these site characteristics. All these data are available for the operator and for the regulatory body.

Q.No	Country	Article	Ref. in National Report
176	Korea, Republic of	Article 17.2	139

Question/ Comment	It seems that an earthquake with a spectrum 1.5 times greater than that of the SSE is defined as a hazard beyond the DBE. Is there any detailed technical basis to determine the value as a spectrum 1.5 times of the SSE?
Answer	All the Hardened Safety Core SSCs have a specific Safe Shutdown Earthquake called SND. The SND is 1.5 times higher than the SSE of the other safety systems of the plant. Note that the SND is defined with the respect of the SSE based on the site seismology (and not

with respect to the Design Basis Earthquake as stated in the question). The 1.5 factor is of the order of magnitude of the margins between the Maximum Historically Probable Earthquake (MHPE) and the SSE. From a physical point of view this corresponds to a potential earthquake of the magnitude of the SSE with a reduced focal distance. A supplementary cross-check was made with respect to the probabilistic seismic hazard assessment so that the SND covers the seismic events with a return period of 20 000 years.
In addition, the seismic probabilistic risk assessment will be done in order to check that the core melt frequency is acceptable.

Q.No	Country	Article	Ref. in National Report
177	Romania	Article 17.2	17.1.1

Question/ Comment: What are the requirements and criteria used for the establishment of the "exclusion area" around nuclear power plants? "Exclusion area" refers to an area surrounding the nuclear power plant, in which the licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area.

Answer: There are no "exclusion areas" around French nuclear installations. Nevertheless, restrictions for the settlement of activities and families around nuclear installations exist.

Q.No	Country	Article	Ref. in National Report
178	Turkey	Article 17.2	138-139

Question/ Comment: It is stated that a deterministic approach is used to define the seismic hazard to be considered in the design of the nuclear facilities. In this context, a further degree of intensity is added to the MHPE to define the SSE. Could you briefly explain the meaning of “further degree of intensity? How is it defined in French regulations?

Answer: For instance, if the Seismic Intensity of the MHPE event is 5, then the seismic intensity of the SSE is 6. The margin between the MHPE and the SSE is +1. This is defined by the ASN Fundamental Safety Rule RFS 2001-01.

Q.No	Country	Article	Ref. in National Report
179	Turkey	Article 17.2	138-139

Question/ Comment: Is there any study or result regarding the seismic margin assessment of the French NPPs? If there is, could you briefly give some information about this?

Answer: Seismic Margin Assessment was performed in the years 2000, for the Tricastin site (900 MW standard design).
Probabilistic Seismic Risk Assessment was performed on the Saint Alban site in the years 2010 (1300 MW standard design).
Both studies indicate that the nuclear standard design induces important levels of margin, mainly by the use of broad band Floor Response Spectra (and by neglecting the favourable effect of the Soil Structure Interactions). These studies revealed some local issues that have been corrected (cable trays friction supports etc.) and a good overall seismic robustness.
The methodology used for SMA and seismic PRA is based on EPRI documents and some improvements have been made and published (see the paper “P755 Determination of probabilistic seismic response spectra …” at SMIRT 21 conference in 2011).

Seismic Margin Assessment was also performed as a part of the design for new reactors (see the paper P258 at the SMIRT 21 conference).
In addition, seismic walk downs have been performed on each NPP since the 1990' in order to identify seismic interactions issues and required retrofitting were made (masonry walls, anchorages of non safety equipments etc.).

Q.No	Country	Article	Ref. in National Report
180	Turkey	Article 17.2	143

Question/ Comment

It is stated that for the sizing of protection against flooding, the sites use basic safety rule RFS I.2.e (issued 1984) which defines a method for determining the water levels to be considered when designing the facilities. This method is based on the defining of the flood safety level and provides for three different cases: sites by the sea, sites on rivers and sites on estuaries.
Could you please give some information on this method regarding the determination of design basis flooding levels? Is there any regulation defining the return period (historical period) of flooding in order to determine the design basis values?

Answer

a) For the three configurations of nuclear sites in France, the Fundamental Safety Rule RFS I-2-e defines the method for determining the reference flood safety level to be considered when designing the facilities.
Riverside sites
The flood safety level is the higher of the two following levels:
- Level reached by a river whose flow is obtained by increasing the thousand year flood level by 15%
- Level reached by the combination of:
o the flood wave resulting from the most penalizing upstream dam failure event,
o and the 100-year river flood (or the highest historical flooding event if this is higher)
Coastal sites
The flood safety level is defined by combining the maximum high tide (coefficient 120) with a 1000-year surge effect (storm surge and barometric effects).
Sites on estuary
The flood safety level is the highest of the three following levels:
- Level reached by the combination of the maximum high-tide (coefficient 120) with a 1000-year river flood.
- Level reached by the combination of the maximum high-tide (coefficient 120) with a 1000-year surge effect (storm surge and barometric effects).
- Level reached by the combination of :
o the average high tide (coefficient 70),
o the flood wave resulting from the most penalizing upstream dam failure event,
o a 100-year river flood (or the highest historical flooding event if this is higher).

	b) There are no regulations which define a minimal return period for the safety flood level to be considered in the design basis. While the RFS 1-2-e was undergoing development, there was an initial objective to consider a 10 000-year flooding event in the basic design. This objective however has not been formalized due to the lack of scientific consensus on suitable methods based on statistic extrapolations to calculate flooding event levels with return periods greater than 1000 years. Instead, the principle of augmenting a 1000 year return period event by a certain percentage (i.e. 15%) or factor has been retained in the RFS. In the guide of ASN relative to the flooding hazards, that has been published in 2013, even if it's not explicitly mentioned, the set of reference flood situations (RFS) to take account for the design have been defined using a common probabilistic target to have a certain homogeneity between all the RFS. In compliance with the international practises, the RFSs should have a probability of exceedance of 10-4 per year, in order of magnitude, and should cover associated uncertainties. Link to the guide: http://www.french-nuclear-safety.fr/index.php/English-version/News-releases/2013/ASN-Guide-relative-to-the-protection-against-external-flooding

Q.No	Country	Article	Ref. in National Report
181	Turkey	Article 17.2	138-139

Question/ Comment	In French regulations, there are two definitions for the deterministic earthquakes, the safe shutdown earthquake (SSE) and "envelope" design-basis earthquake (DBE) which is taken into consideration in the standardized plant series for the design of the nuclear island. What are the main differences between SSE and DBE from the point of their characteristic?
Answer	The Safe Shutdown Earthquake (SSE) is defined by the seismological characteristics of each site according to the French Fundamental Safety Rule RFS 2001-01. The Design Basis Earthquake (DBE) is defined by the licensee for industrial reasons. The safety requirement is that the DBE shall cover the SSE for all NPP of a given technical standard in the range of interest of safety structures and components. For technical and economical reasons, the DBE covers the SSE with more margins in order to have a standard design accommodating several sites and to provide margins in the case of future reassessment of the SSE because of new seismological knowledge This is generally achieved by defining the DBE as a broad band spectrum (EUR or NUREG) pinned at a relatively high value of PGA (0.15 – 0.25g) given the seismological context.

Q.No	Country	Article	Ref. in National Report
182	Germany	Article 17.3	17.2.1. page 140

Question/	The "seismic interaction" procedure aims to prevent equipment that must remain operational after an earthquake from being damaged

Comment	by non-seismic classified equipment or structures. […] A guide to the management of the seismic interaction hazard on the NPPs currently being drafted will define the organisational measures to implement on the sites and specify the roles and responsibilities of the protagonists and the prevention measures to implement. […] Please provide further clarification on this specific guide, particularly regarding prevention measures.
Answer	The main prevention measure requested by the guide, requires an assessment of seismic interaction risk, in the risk analysis provided for each operating activity. For that purpose, special rules have been promulgated in order to characterize the seismic interaction risk : for example, the duration of the activity providing the risk, the proximity between the equipment that aggress and the seismic qualified equipment, the ability of the equipment to aggress and damage the seismic qualified equipment, etc...In case of proven risk, correction measures have to be taken : for example, protection of the qualified equipment, fixing the equipment that can aggress, etc...

Q.No	Country	Article	Ref. in National Report
183	Turkey	Article 17.4	148

Question/ Comment	After Blayais Flooding accident (1999) it is pointed out that, In 2013 ASN published a new guide on protection against the external flood risk for nuclear facilities. What are the 11 different hazards taken into consideration particularly in hydrology and meteorology?
Answer	The first step of the hazard characterization is to list the water sources that could initiate or contribute to a flood affecting the site in question: - rainfall - groundwater - seas and oceans - watercourses (river and canals) - natural reservoirs (lakes, glaciers) - man made reservoirs (storage dams, tanks, water towers, pipes, etc). The second step in the approach is to list the events or combinations of events that could cause a flood hazard for the installation in question, for each of the identified water sources. A particular "event" is usually characterised by a physical quantity that defines its intensity (volume, height, flow rate, etc.) and if applicable, a probability or frequency of exceedance of that intensity, and a duration. Finally, a "Reference Flood Situation" (RFS) is defined on the basis of an event or a combination of events whose characteristics may be increased if necessary (unfavourable combination or margin to compensate for the limits of current knowledge). These RFS are determined on the basis of a statistic or deterministic study. The guide lists 11 RFS: - Local rainfall

- Small watershed flooding
- Large watershed flooding
- Deterioration or malfunctioning of structures, circuits or equipment
- Mechanically induced wave – Malfunctioning of hydraulic structures (A mechanically induced wave is a wave travelling along the open surface of water in a channel, induced by a sudden variation in the speed (flow rate) of the flow.)
- High groundwater level
- Failure of a water-retaining structure
- Local wind waves
- Sea level
- Ocean waves
- Seiche (A seiche is a stationary wave that can occur in a closed or semi-closed area of water such as a harbour, pond, lake or bay. In a semi-closed maritime dock, seiches are caused by the penetration of long waves coming from the open sea.)

Q.No	Country	Article	Ref. in National Report
184	Turkey	Article 17.4	148

Question/ Comment	Following the flooding of the Blayais site in 1999, a new concept called "Protective Volume Perimeter" on all the sites has been introduced. Is this applicable to every existing site?

Answer

EDF protected sensitive rooms by the definition of a compact watertight area: the openings in the outer walls of the buildings and rooms containing equipments important for safety were closed off to prevent the entry of water into these rooms.

EDF took provisions to prevent water ingress into the watertight area:
- Identification of all water paths
- Qualification of the material used to plug them
- Prevention of potential ways of by passing this area
- Provisions to close them quickly if opened
+ Provisions to cope with a residual leakage (mobile pumps; detection of water)

This leaktight envelop is applicable on each site and is specific to each existing site. This protective volume perimeter depends on the sites vulnerabilities, including the superstructures up to the required protection level.

The concept called "Protective Volume Perimeter" is also translated by “Watertight Volume Perimeter” in the guide n°13 of ASN which details the recommendations concerning the external flooding hazards. Since 2007, these recommendations have first been implemented on Nuclear Power Plants (NPP) in the scope of the Blayais NPP experience feedback method drawn up by EDF. The French regulations require that the flooding hazard has to be taken into consideration in the demonstration of nuclear safety of basic nuclear installations (BNI). Concerning the non NPP installations, it has been decided that the recommendations formulated in the guide will systematically be implemented at the design stage of new installations (e.g. ASTRID).For the existing installations or these which are currently being built (e.g. ITER), the Protective Volume Perimeter will be taken into account at next periodic safety review, as all the recommendations of the guide. Among the existing BNI, some works have recently been realised on the experimental reactor CABRI in the prospect of the re-commissioning of the modified installation : particular attention has been paid to openings (doors,…) that could allow water to enter buildings. In addition, water entry channels situated below the platform setting level are closed off as required so that in a flood situation, the water cannot reach the rooms housing important protection elements associated with nuclear safety. The design of the doors take into account the hydraulic pressure associated with the potential presence of water outside the Protective Volume Perimeter. Passive measures are favoured, so that there is no need for early human intervention to close these water entry pathways in the event of flooding of the site.

Q.No	Country	Article	Ref. in National Report
185	Finland	Article 18.1	chapter 18.3.2.2

Question/ Comment: It is stated that no nuclear reactor apart from Flamanville-3 EPR has an alternate heat sink. Has ASN considered to require operator to study possibilities to introduce an alternate heat sink also to the other reactors?

Answer: After Fukushima, as part of the hardened safety core, an alternate heat sink will be implemented on each NPP (one for each reactor/spent fuel pool).

Q.No	Country	Article	Ref. in National Report
186	Japan	Article 18.1	p159

Question/ Comment: French report refers two ultimate back-up electric sources, one is a power source provided by a turbine generator and the other is ultimate back-up diesel generator. Please clarify those electric sources. Are those two are both required for a reactor or a nuclear power site with multi reactors?

Answer: Since the end of the 1980's, each EDF NPP's reactor has an emergency turbine driven generator which uses the steam produced by steam generators to cope with SBO situations (along with EFWS turbine driven pumps to feed the SGs). Nevertheless, its power capability is not sufficient to provide power to all the equipments required to cope with extreme situations postulated in the stress tests, and in the following ASN requirements.
In order to deal with these extreme situations, EDF plans to implement an additional electric power source which will be a new ultimate EDG.

Q.No	Country	Article	Ref. in National Report
187	Korea, Republic of	Article 18.1	152

Question/ Comment	In Safety Standards No. SSR-2/1, IAEA described that Design Extension Conditions(DEC) shall be considered for the design of a new NPP. Does France have a plan to reflect DEC into regulation? If so, please explain the plan.
Answer	SSR-2/1 DEC encompasses what was previously considered BDBA, i.e. severe accidents as well as multiple failures events. French Ministerial Order of 7 February 2012 does not discriminate between DBA and DEC. The order requires the licensee to assess single initiating events as well as single external or internal hazard (including consequential faults) as well as plausible combination of initiating events or hazards. Plausible combinations are to be identified via PSA input, operating experience feedback (at the licensee own facilities but also in other relevant facilities, whether national or foreign) and R&D. Plausible combinations directly address the issue of potential multiple failures. Severe accident are however not explicitly mentioned as the order is applicable to all nuclear installations (including those were no nuclear fuel is present) but is nevertheless covered by the need to consider within OEF the severe accidents which have already occurred in the world.. ASN is currently developing a guide on the design of NPP which will explicitly include severe accidents.

Q.No	Country	Article	Ref. in National Report
188	Korea, Republic of	Article 18.1	155

Question/ Comment	In the report for Article 18, EDF has proposed installation of ¡°Ultimate back-up diesel generator¡± by 2018 for loss of electrical power supplies to the NPP. Does the ¡°Ultimate back-up diesel generator¡± have different design requirements from those of the existing EDGs? If so, please describe the differences.
Answer	The new ultimate EDG is designed to resist more severe hazards (earthquake, tornado, flooding) than the existing EDGs. The ultimate EDG is part of the "hardened safety core" SSCs. Regarding safety classification : the components of the "hardened safety core" are considered as important to safety and assigned to the so called "IPS-NC" classification, which corresponds to the third level in the international safety classification system (IAEA Guide referenced DS367). The SSC of the "hardened safety core" are subjected to the following requirements : - aptitude to carry out the needed mission in extreme accidental conditions - design and manufacture under quality assurance for the new SSC, - aptitude for periodic tests, - follow-up throughout all the installation lifetime. All the Hardened Safety Core SSCs have a specific Safe Shutdown Earthquake called SND. The SND is 1.5 times higher than the SSE of the other safety systems of the plant. Note that the SND is defined with the respect of the SSE based on the site seismology. The 1.5 factor is of the order of magnitude of the margins between the Maximum Historically Probable Earthquake (MHPE) and the SSE. From a physical point of view this corresponds to a potential earthquake of the magnitude of the SSE with a reduced focal distance.

Q.No	Country	Article	Ref. in National Report
189	Pakistan	Article 18.1	Section 18.3.4.1.2 , Page 164

Question/ Comment: Reference section 18.3.4.1.2, France may like to share the root causes of numerous defects discovered in the welds of support brackets of the reactor building polar crane?

Answer: The defects detected in the supports brackets of the reactor building polar crane were due to the combination of the following causes :
- A complex design with numerous full-penetration welds and subsequent difficulties in the manufacturing and in the control of the brackets;
- An insufficient preparation of the welders for this type of parts leading to weld defects;
- Surveillance and controls arrangements which didn't allow the detection of the defects.

Q.No	Country	Article	Ref. in National Report
190	Switzerland	Article 18.1	18.1.4.1.1. P, 153

Question/ Comment: One of the major tasks regarding enhancement of the nuclear safety of the French NPPs is the implementation of a "hardened safety core" for each plant. Could France please provide information on the main design criteria to be used for those systems, especially regarding redundancy, autarky, seismic robustness and type of the diverse UHS to be used? For sites with no ground water available, how would a diverse UHS be realized? What is the period of time for these backfitting measures?

Answer: a) In case of extreme situation (LUHS + SBO superposition), potentially induced by an earthquake or a flood significantly exceeding the design level, the hardened safety core is defined to prevent large radioactive releases with long term consequences on the environment. It is an ultimate line of defence for extreme, highly improbable situations.
The hardened safety core have to be :
- composed of a limited number of Systems, Structures and Components (reliability),
- protected against extreme earthquake, flood and tornado, explosion, lightning, extreme climatic conditions, wind, snow, accidental rain, hail storm, wind generated missiles...
- protected against the effects that could be induced by these hazards,
- operable even if all other components are out of service (e.g. dedicated electrical source and I&C),
- operable without any material or human support from the outside during 24 hours following the event until FARN set-up (Nuclear Rapid Intervention Force),

There is no redundancy requirement for the components of the hardened safety core, but it shall be possible to carry out their function by alternative means (provided by the ""FARN"" after 24h) if maintenance is required during the operation of the hardened safety core.
b) Groundwater pumping is the technical device privileged for the ultimate source of water. However for sites with no ground water available, existing storage (plant water supply basins) or new dedicated basins will be used. The design of these storages makes it

possible to guarantee a sufficient autonomy until the deployment by the ""FARN"" of mobile means of re-feeding.
c) The back fitting measures following the stress tests will be implemented according to a 3 step program:
-a first step has already begun in 2012 and will be completed by 2015: short term improvement of SBO and LUHS management by implementation of mainly mobile SSC (pumps, EDG, multiple connexions to the existing systems...) and of the Nuclear Rapid Action Force (FARN in French acronym)
-a second step by 2018-2020: implementation on each unit of the main ""hardened safety core"" SSCs designed to meet the Fukushima accident feedback of experience (new fixed EDG and new water source for each unit, a new Emergency Control Centre for each site, reinforcement of the plant staff...)
-a third step to meet the safety objectives related to life extension, implemented in the frame of the next PSR (reduction of core damage frequency, reduction of the risk of large radioactive releases to the environment.

Q.No	Country	Article	Ref. in National Report
191	United Arab Emirates	Article 18.1	156

Question/ Comment: The National Report states that “in October 2012, ASN completed its conformity assessment of the first newly manufactured steam generators. Further to this assessment, ASN considers that AREVA has acceptably demonstrated the conformity of the equipment with the provisions of the ESPN order, thanks to the efforts made during 2012. ASN does however consider that AREVA’s practices must evolve even further to fully adapt to the requirements introduced by the regulations.” Could France elaborate further on this matter, in particular what are the challenges in meeting the new regulations in this respect?

Answer: To take into account the provisions related to the conformity assessment of equipment, ASN has made doctrinal elements available for manufacturers to enable them to draw up the technical documentation. It concerns several items like risk analysis, justification of compliance with requirements for materials, capability to perform inspection, notice of instructions, final visual examinations.

Q.No	Country	Article	Ref. in National Report
192	China	Article 18.2	section 18.2.2

Question/ Comment: Description In section 18.2.2 ¡°in 2011 and 2012 concerned more than 400 suppliers out of 866 who were subject to a F3 manufacturing monitoring plan ¡±
Question: How to guarantee the qualification of these supervision personnel? For the rest of the other suppliers, how to conduct monitoring or supervisions during equipment manufacturing?

Answer: As a whole, the in-factory monitoring of the manufacture of mechanical and electrical equipment in 2011 and 2012 dealt with more than 400 suppliers out of a panel of about 866 who were subject to a Flamanville-3 manufacturing monitoring plan.
Question 1 :
Performing manufacturing surveillance requires skills and techniques requiring long period of training or of job companionship with other skilled inspectors. EDF requirements for its inspectors is the award of a Special Initial Inspection Program accreditation

(minimum 6 months training duration) + attendance of refresher courses + companionship with other skilled inspectors.
Question 2 :
On FA3 but also other EDF manufacturing monitoring plans, a graded approach is applied on documentation and manufacturing surveillance. Among several criteria, nuclear safety is one key factor leading to intensive inspection activities.
On the contrary, non safety classified equipments (and the suppliers performing these manufacturing) get lower level of surveillance on a risk-informed approach. Thus the whole supply chain involved in this kind of procurement may not be subject to surveillance.
Anyway such supply chain may be assessed through other processes (QA Audits) in order to provide a satisfactory level of confidence in its capabilities

Q.No	Country	Article	Ref. in National Report
193	Korea, Republic of	Article 18.2	4.2.1, 157

Question/ Comment	The regular contact between the ASN and STUK is mentioned in the National Report regarding the construction and installation of EPR-type reactors in France and Finland. Is there any similar contact between the ASN and the Chinese regulatory body NNSA since the EPR technology is also applied for the Chinese project at Taishan? If so, please provide some information.
Answer	In the framework of the bilateral cooperation agreement between ASN and NNSA, several meetings are held each year, not all devoted to EPR. In June 2013, was organised, on behalf of MDEP, a workshop dedicated to the commissioning tests for EPR; This workshop gathered many regulatory bodies' representatives interested in this topic. Actually, because of the distance between China and France, there are not so many visits of the respective construction sites.

Q.No	Country	Article	Ref. in National Report
194	China	Article 18.3	section 18.3.2.2

Question/ Comment	Description In section 18.3.3.2 ¡°on each reactor in service, one ultimate electrical power source provided by a turbine generator driven by steam from the steam generators.¡± Question: For the reactor which ultimate electrical power source provided by a diesel generator, is it necessary to replace it with a turbine generator? How to solve the spare part supply problem?
Answer	The steam driven turbine driven generator (see below) and the ultimate diesel generator are two separated systems. EDF has not studied so far any replacement of existing electricity supplies by the new ones, implemented as part of the "hardened safety core". Since the end of the 1980's, each EDF NPP's reactor has an emergency turbine driven generator which uses the steam produced by steam generators to cope with SBO situations (along with EFWS turbine driven pumps to feed the SGs). Nevertheless, its power capability is not sufficient to provide power to all the equipments required to cope with extreme situations postulated in the stress tests, and in the following ASN requirements.

In order to deal with these extreme situations, EDF plans to implement an additional electric power source which will be a new ultimate EDG.

Q.No	Country	Article	Ref. in National Report
195	Germany	Article 18.3	18.4.2.2. p.157

Question/ Comment

ASN validation of the instrumentation and control architecture:
[…] In response to the ASN request in a letter dated 9th July 2010, EDF presented alternative design measures to those initially envisaged. These new design measures consist more particularly in grouping within a "hardened safety core" system certain safety functions hitherto not installed on the Teleperm XS platform. These measures make it possible to deal with total loss of the SPPA T2000 platform combined with certain accident situations. […]
Further to the analysis of these modifications carried out by IRSN and the opinion of the Advisory Committee for nuclear reactors (GPR) returned on 16th June 2011, ASN considers that the I&C architecture of the Flamanville-3 EPR reactor proposed by EDF is able to guarantee the safety of the systems used to manage incident or accident situations and their independence from the control systems used to run the facility in normal operating conditions. EDF can thus continue to deploy this system, for which the detailed design will be analysed by ASN prior to the commissioning authorisation.
Please give further details on the digital I&C.
Does ASN consider SPPA T2000 as diverse to the Teleperm XS platform?
On which platform is the hardened safety core system installed? Which functions are installed in the hardened safety core system and by which features was the system “hardened”?

Answer

Does ASN consider SPPA T2000 as diverse to the Teleperm XS platform ?
Diversity between the two platforms has indeed been assessed and considered acceptable.

On which platform is the hardened safety core system installed ?
The I&C “hardened safety core” system is installed on the TXS platform

Which functions are installed in the hardened safety core system and by which features was the system “hardened”?
The previous assessment of SPPA T2000 performed prior to 2010 could not conclude on the capability of this platform to satisfy the requirements for F1B safety classified function.
Hence, ASN requested EDF to find another design solution to implement the F1B function that was intended to be installed on SPPA T2000.
EDF proposed then to add a “hardened safety core” of F1B safety functions, implemented on the TXS platform (“hardened” meaning “fully compliant” with all safety requirements for such F1B functions as opposed to their “degraded” version implemented on the

SPPA T2000), in order to be able to cope with accidents and the total loss of the SPPA T2000.
At the same time, a detailed assessment of one specific automaton of the SPPA T2000 finally reached the conclusion that this automaton was able to satisfy all safety requirements for implementation of F1B functions.
As a conclusion, now, thanks to the availability of “true” F1B functions on the SPPA T2000, the “hardened safety core” is no more required for the safety demonstration.
However, EDF decided to keep it as a complementary I&C robust feature.

Q.No	Country	Article	Ref. in National Report
196	Switzerland	Article 18.3	18.3.2.2. P, 160

Question/ Comment: Regarding loss of cooling systems or ultimate heat sink (UHS), ASN states that the reactors in service are designed to have an autonomy of at least 100 hours after a heat sink loss. The Swiss analyses regarding core cooling after SBO with consequent failure of the UHS did show that dryout of the steam generators (SG) occurs within approximately 1 hour after the loss of feedwater, with the urgent need of feeding the SGs by accident management measures. Could you please elaborate on the means which are provided in the French NPPs to assure an autonomy of more than 100 hours in case of SBO/loss of UHS?

Answer: In the case of Station Black Out, the emergency feed water (EFW) stream driven pump (s) is (or are for 4 loops design) still running. EFW tank is refilled using all available site make-up sources, allowing the evacuation of the residual power for 100h. If make-up sources to the EFW tank are not sufficient, or not available, the residual power is evacuated for ~8 to 10 hours using only the EFW tank water. During this delay, some emergency supplies (e.g. using raw water) can be used to refill the EFW tank.
In the extreme case of SBO and at the same time loss of the EFW steam driven pump (s), as studied in the stress tests, SGs can be dried out in nearly 1 hour. EDF has decided to add an ""ultimate SG make-up system"" : a new motor driven pump powered by an ultimate emergency diesel generator automatically feeds the SG with water provided by a new specific tank . The tank is refilled by an ultimate water source, either from pumping of underground water or by a raw water storage. All these SSC belong to the ""hardened safety core""

Q.No	Country	Article	Ref. in National Report
197	Switzerland	Article 18.3	18.3.2.2. P, 159

Question/ Comment: In this section of the national report, France states that for each reactor in service, one ultimate electrical power source provided by a turbine generator driven by steam from the steam generators will be backfitted. Is this an additional measure to the "hardened safety core" or will this steam driven turbine generator be part of the hardened core?

Answer: The existing emergency turbine generator driven by the steam produced by steam generators is not part of the "hardened safety core". It has been implemented just after the initial design to face less severe SBO situations than the ones postulated in the stress tests and by the ASN 2012 requirements.

Q.No	Country	Article	Ref. in National Report

198	Ireland	Article 19.1	p 166

Question/ Comment: It is noted that for new build nuclear power plants, the licensee is required to provide a number of different documents and procedures at least one year in advance of commissioning – this includes the on-site emergency plan (PUI). The PUI is assessed by a committee of experts – how does the requirement to have a test of the plan in advance of commissioning (Article 16) factored into this assessment?

Answer: The licensee is requested to provide the PUI in advance of the commissioning. Then it is assessed. One specificity of the French NPPs is that they are standardized. After the commissioning of the NPP, the PUI will be tested annually by the operator and also during the regular national exercises. There are around 12 exercises each year.

Q.No	Country	Article	Ref. in National Report
199	Switzerland	Article 19.1	Section 19

Question/ Comment: Approximately how many inspections have been performed by the Regulatory Body in each NPP in respect to the national investigations and actions taken in the light of the Fukushima Daiichi Accident?

Answer: One or two "targeted inspections" conducted in 2011 and one follow-up inspection (in 2012) have been performed in each NPP to take into account accident at Fukushima Daiichi NPP.

Q.No	Country	Article	Ref. in National Report
200	Switzerland	Article 19.1	P, 166

Question/ Comment: Is there a Regulatory Body requirement/regulation for assistance to an accidently affected NPP site/area from outside the affected area? Would such assistance from other parts of the country cover hardware and personal? Is air-borne assistance arranged?

Answer: After the Complementary safety assessments (CSAs), in 2012, ASN requested the licensees, in particular EDF which has a standardized fleet of NPPs, to establish complete teams able to provide assistance to an accident site. Thus, the licensee shall present ASN with the measures it intends to take in order to provide specialised teams capable of relieving the shift teams and deploying emergency response resources in less than 24 hours, with operations starting on the site within 12 hours following their mobilisation. This system may be common to several of the licensee’s nuclear sites. The team size shall be fixed to enable them to respond to all the reactors of the site and have measuring instruments that can be deployed at their arrival. The licensee shall specify the organisation and size of these teams, in particular: the activation criteria, the tasks incumbent upon the teams, the material and human resources at their disposal, the individual protection equipment, the system put into place to ensure the maintenance of these material resources and their permanent operability and availability; the training of their staff and the skills currency process. The system shall be able to intervene simultaneously on all reactors of the site by the end of 2014. The licensee shall also present the measures for adaptation of the system to simultaneous intervention
on several of its nuclear sites (see on the ASN website the resolution of 26 June 2012, [EDF-CAT-26][ECS-36]).

Q.No	Country	Article	Ref. in National Report

201	Belgium	Article 19.2	chap. F, sec. 19.2.1, page 168

Question/ Comment

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§ 19.2.1 of the French National Report provides an explanation on the procedure to be used when EDF have to diverge from the normal operating procedures dictated by the STEs. It is mentioned that EDF "must report any such divergence to ASN. ASN examines the temporary modifications to the STEs and may approve them".

Are there explicit and well known criteria elaborated to decide which modifications have to be approved by ASN and which are the modifications that require no formal approval by ASN? In the case of formal approval, must ASN give its approval before the modification may be put into force by EDF?

Answer

All modifications to the STEs are subject to ASN approval before being implemented by EDF.

Q.No	Country	Article	Ref. in National Report
202	China	Article 19.2	section 19.2.3

Question/ Comment

Description In section 19.2.3 ¡°The most frequent deviations result from human failings or the operating organization.¡±

Question:

What¡¯s the style of human factor failure and the operating organization?

In order to decrease the human errors, what measures have been taken?

Answer

For example, EDF has worked to decrease the number of exit of the operation range. In this field, EDF has detailed the requirements for control-room oversight and for conducting sensible transients. Moreover, meetings have been organized by corporate level with members of each NPP to share practices identified by each NPP after analysing technical Specifications non compliance events.

Q.No	Country	Article	Ref. in National Report
203	China	Article 19.3	section 19.3.4

Question/ Comment

Description In section 19.3.4 ¡°The increase results notably from the combined effects of the deformation of the concrete and the loss of the prestressing in certain cables.¡±

Question:

Has the situation increase in the leak rates of the inner wall of some of containments been expected before?

How to control the deformation of the concrete and the loss of the prestressing in certain cables?

Answer

The deformation and the loss of prestressing were taken into account in the design of the 1300 and 1450 MWe containment buildings. However, an increase in the leak rate higher than expected has been highlighted during the ten-yearly periodic tests under pressure of the containment buildings over the last 15 years (and this after the first or second ten-yearly outages). EDF applies the following strategy in order to fulfil the safety criteria: every ten years a study is carried out to know the behaviour of the containment building from a leak tightness point of view. This study is based on the deformations measured on the building and extrapolated up to 60 years

	through Eurocode's laws for creep and shrinkage of the concrete and including the effect of the relaxation of the prestressing cables. If the study leads to the need of repairs to improve the leaktighness (resin coating), works are carried out before the next Integrated Leak Rate Test (ILRT). Concrete deformation and loss of prestressing : The deformation of the concrete (creep and shrinkage) and the relaxation of the prestressing cables are measured through the use of extensometers embedded in the concrete prestressed wall and through dynamometers measuring the loss of tension on some dedicated cables which are unbounded and injected with grease.

Q.No	Country	Article	Ref. in National Report
204	Pakistan	Article 19.3	19.3.4.1.2, Page 173

Question/ Comment	Reference Section 19.3.4.1.2, France may like to share what measures are being considered by EDF in response to the shortcomings observed by ASN on implementation of maintenance methods.
Answer	Maintenance requirements are changing to include operating experience and reliability information on components. Integration of new maintenance requirements is monitored for each plant in order to identify weaknesses in this process and define accurate action plans. This includes optimization of integration delays according to required first maintenance realization and plant outage program. This includes human resources adaptation. Redaction of maintenance requirements program by components and maintenance experts are also challenged in order to reduce the impact for teams, which are in charge of integration. Regarding quality of maintenance work performance, a major project has been launched by EDF in collaboration with major contractors. Improvement of files and risk assessments produced prior to maintenance work is identified as a key factor. Operating experience information is collected and several tools have been implemented and monitored through processes to revised files and risks assessments accordingly. The objective is to share technical facts with contractors before, during and after the maintenance work. This should allow to highlight key information in files and to focus supervision and monitoring of contractors on main critical phases during a maintenance work.

Q.No	Country	Article	Ref. in National Report
205	Romania	Article 19.3	19.3.4.1.2 Maintenance activities

Question/ Comment	In Section 19.3.4.1.2 Maintenance activities, on page 173, it is mentioned that ASN has observed certain recurrent shortcomings in relation to maintenance. Were these non-compliances with regulatory requirements or just deviations from expected standards of performance? What regulatory actions does ASN take to direct the licensees to improve their performance in this area?
Answer	These non-compliances were deviations from expected standards of performance. The main cause of these recurrent shortcomings in relation to maintenance are often linked with a lack of technical skills by the newest workers (there are a lot of EDF workers are closed to be retired), bad instructions in the documents used for maintenance activities or lack of time allowed to perform these activities (often practiced during shutdowns for core-reactor refuelling). The licensee took measures to avoid these shortcomings. ASN inspects the licensee to verify that workers get trainings, the licensee planned the maintenance of the NPP with sufficient time... ASN considers

that the shortcomings in relation to maintenance will still be a challenge for next years because EDF needs to hire a lot of worker superseding whose will be retired.
The regulatory basis used for these inspections are in the ministerial order from February, 7th 2012 which force the licensees to prepare the activity, to check it, and to oversight the subcontractors who have done the activity in the aim to improve the quality of the activities here above.
However, there are no indicators or criteria introduced in the regulatory basis and used for inspecting the licensee.

Q.No	Country	Article	Ref. in National Report
206	Romania	Article 19.3	19.3.4.1.3 Condition of equipment, p.174

Question/ Comment	What are the categories of maintenance activities of which ASN requires to be informed by the licensees? What types of maintenance activities do the ASN inspectors typically observe?

Answer

What are the categories of maintenance activities of witch ASN requires to be informed by the licensees ?
On a general basis, the programme of maintenance planned by EDF during the reactor outages (including in service inspection) is reviewed by ASN before the beginning of the outage. Before operation start-up, the license has to send to the ASN an assessment describing all maintenance activities done during the outage and the results obtained. On, that basis, ASN grants or refuse the license for start up.

For nuclear pressurized equipments (regulated by the interministerial order of 10th November 1999 relative to the monitoring of operations of the main primary system and the main secondary system of pressurized water nuclear reactors and/or the ministerial order of 12th December 2005 relative to nuclear pressure equipment), in-service inspection programme is approved on an individual basis by ASN.
For the main primary system and the main secondary system, rules on the subject are defined by the order of 10th November 1999. The standard maintenance programs are approved by ASN. The licensees are requested to inform the ASN for any maintenance activities on the main primary and secondary systems of PWRs. In addition to this, a licensee planning a significant maintenance or repair on an equipment of the main systems has to obtain an ASN agreement. What should be considered as significant has been defined by regulation; it includes for example steam generator
replacement, pressurizer heater replacement, steam generator chemical cleaning, relief valve flange seal resurfacing, latch mechanism replacement, pressurizer high pressure cleaning.
For nuclear pressure equipments other than main primary and secondary systems of PWRs, rules are defined by the order of 12th December 2005. No need of ASN information is requested for
maintenance activities by this order. Nevertheless, for those equipments, significant maintenance or repairs have to be endorsed by an notified body agreed by ASN. In this case, what should be

considered as significant has been defined by a professional guide (that guide being reviewed by ASN).

What types of maintenance activities do the ASN inspectors typically observe ?
The ASN inspectors are involved in assessment described before for main primary and secondary systems of PWRs. During outages they observe all kind of maintenance activities, depending on
the programme, and taking into account the safety significance of the operation as well as the operating feedback.

Q.No	Country	Article	Ref. in National Report
207	Russian Federation	Article 19.3	Subsection 19.3.4.1.2

Question/ Comment This Subsection states that in 2010 EDF informed ASN on its intention to use a new maintenance approach known as AP913, which through monitoring will allow initiating relevant repairs before an equipment failure take place.
Please, provide more details on the AP913 approach. Who did develop it? Was this approached used before; if yes, what were the outcomes?

Answer Advanced Process AP-913 was proposed by INPO before 2000. In November 2001, INPO issued Rev. 01 – AP-913 “Equipment Reliability Process Description” which contains main information about this methodology. The purpose of AP-913 was to describe an equipment reliability process to ensure that all utilities would maintain high levels of safe and reliable plant operation in an efficient manner. The equipment reliability process was designed with the direct participation of several utilities actively involved in improving and reengineering their own processes. The main steps of the methodology require to scope and identify critical components (regarding criteria proposed by the methodology), monitor performance of components, implement corrective actions (including corrective maintenance or revision of preventive maintenance). At EDF this is the first implementation and evaluation of effective outcomes is expected in next years. AP-913 is implemented for several years in US companies (Constellation, Duke Power, Exelon) and members of the Equipment Reliability Working Group share annually their experience and effective outcomes, provided by the methodology.

Q.No	Country	Article	Ref. in National Report
208	Germany	Article 19.4	19.4.2, page 176

Question/ Comment Risk of slow pressurisation of the containment:
On the reactor fleet, the time before containment is lost due to exceeding of the containment mechanical characteristics varies from one to several days depending on the assumptions adopted for the studies. EDF considers that this leaves the operator the time to take steps to avoid containment destruction while optimising control of radioactive releases. This risk is countered by a containment venting-filtration system and the associated operating procedure that preserves its long-term integrity. This system opens after 24 hours as from a minimum pressure equal to the containment design pressure. In response to the ASN requirement, EDF has started to study the possibilities of improving this venting-filtration system, including a review of the hydrogen risk and its potential

	consequences, and the resistance to earthquakes. The results will be submitted by the end of 2013. […] Please elaborate on the results of the study and notably on the possibilities of improving the venting- system.
Answer	First in the frame of the Plant Life Extension Program, and second in response to ASN requirements after the Fukushima accident EDF has studied: -the strategy to ensure cooling of the core in the vessel or in the reactor pit in case of SA; -the interest and feasibility of improving the existing Filtered Containment Venting System (FCVS). EDF position has been submitted to ASN at the end of 2013. The technical solution proposed by EDF is to keep a closed containment even in case of SA including a core melt, for limiting as much as possible radiological releases to the environment. To achieve this, EDF has decided to implement, as part of the “hardened safety core"", new dedicated SSCs to remove the decay heat from the containment building with a recirculation loop cooled by a dedicated heat sink. In addition, in the frame of the “defence in depth” concept, to ensure an ultimate protection of the containment, and taking into account the Fukushima experience, seismic resistance of the existing FCVS installed on the French plants will be increased. As the FCVS doesn’t belong to the “hardened safety core"", the level of seismic resistance taken into account will be less than for the hardened safety core SSCs Moreover, in order to strongly limit gaseous iodine production, STB (Soda Tetra borate-Borax) baskets will be installed in the containment sumps of 1300 MWe and N4 NPPs which are not equipped only with Silver Indium-Cadmium control rods. Improvement of the filtration capacity of FCVS, particularly for organic iodine, has been studied but, according to the above decisions, the installation of such improved FCVS is not necessary. Concerning hydrogen combustion risk at the opening of FCVS: - Since the original design of FCVS, a preheating system is installed in order to avoid, by steam condensation, combustion of hydrogen in the system. Note: installation of PARs (Passive Autocatalytic Recombiners) in all French NPPs has very strongly decreased the occurrence of this potential risk, but, using deterministic considerations, it cannot be totally excluded in the case of accidental situations with late loss of core water injection leading to insufficient delay for sufficient hydrogen recombination before FCVS opening. - After recombination of all oxygen of the containment building by PARs, which could occur only in case of MCCI, there is a risk of combustion between hydrogen (and carbon monoxide) coming from the RB and oxygen initially present in the FCVS. This is a very transient risk. This situation is studied in the frame of the Plants Life Extension Program and more especially, in the field of the measures to ensure cooling of the core in the vessel or in the reactor pit which will strongly limit the production of combustible gases during MCCI and consequently will eliminate this kind of risk.

Q.No	Country	Article	Ref. in National Report
209	Germany	Article 19.4	19.4.2. page 177

Question/ Comment	[…] In a long-duration total loss of electrical power supply situation combined with the loss of water supply to the steam generators, none of the present injection means would allow flooding of the corium in the vessel and in the reactor pit. Consequently, EDF envisages, for the reactor fleet, using a generator driven pump supplied by the Ultimate Backup Diesel, allowing injection of water into the primary cooling system. […] Please provide further clarification on the concept of water injection into the primary cooling system during a severe accident.
Answer	In case of long-duration total loss of electrical power, combined with the loss of water supply to the steam generators, there is a risk to have a severe accident with core melt. To mitigate these situations, water injection in the primary cooling system will be enabled with a dedicated generator driven pump powered by the Ultimate Emergency Diesel, pumping water from the RWS tank or from the in-containment sumps in a recirculation mode. This pump will be designed to operate in severe accident conditions and to inject water in the vessel in a feed and bleed mode after opening of the pressurizer relief valves. The injection of water in the primary cooling system after the loss of water supply to the steam generators is likely to avoid core degradation or to stop the core degradation if operated early enough after the onset of the severe accident. In the worst case for which the corium would be relocated in the vessel lower plenum, leading to the vessel failure, this pump would allow the corium to be flooded in the reactor pit. The water injected in the primary cooling system would then fall into the reactor pit after the vessel failure. It could then be possible to stabilize the corium in the reactor pit during the ex-vessel phase (core concrete interaction with water injection). Studies are in progress at EDF to assess the efficiency of corium stabilization by flooding of the reactor cavity after vessel failure . This kind of solution will be optimized to significantly reduce the risk of basemat melt-through in case of severe accident with failure of the vessel. In this context, the implementation of a spreading area in a room neighbouring the reactor cavity could be considered if cooling of the corium in the cavity would not be sufficient. These studies take into account the results of international R&D dedicated to molten core concrete interaction with flooding of the corium (MACE and CCI-OECD programs).

Q.No	Country	Article	Ref. in National Report
210	Germany	Article 19.4	19.4.2. p. 178 – 179

Question/ Comment	Habitability of the control room EDF's preliminary studies on the habitability of the control room in the event of a severe accident lead to envisaging not having operators permanently present in the control rooms in the period following opening of the venting-filtration system and maintaining the supervision and monitoring of the facilities by complementary measures. […] Please elaborate how further measures are envisaged to be realized (for example closing of the venting- line), in case that the control room is not habitable.
Answer	In the stress tests, it has been postulated that all the electrical supplies were lost. So, the control room ventilation and filtration system

is not available. The leaks of the double wall containment are not filtrated either, and the pre-heating of the venting line of the containment is also unavailable, inducing a potential risk of H2 explosion. In such conditions operators couldn’t stay permanently in the control room during the period following the opening of the FCVS system.

Following the stress tests, it was consequently decided to implement a small fixed diesel generator to supply ventilation and filtration of both the control room and of the double wall containment annulus, and also a mobile diesel generator brought by the FARN to supply the FCVS pre-heating. The “bunkered” ultimate emergency diesel generator (belonging to the hardened safety core) will also supply these systems when installed.

Moreover, on four loops plant, baskets of soda tetra borate will be implemented to maintain basic the water in the sumps in order to limit organic iodine releases.

Thanks to these modifications, the control room has not to be evacuated, even temporarily, after a severe accident.

Eventually, EDF has decided to implement, as part of the “hardened safety core"", a new dedicated system to remove the decay heat from the containment building in case of severe accident without venting of the containment. The radioactive releases and dosimetry on the site will be still more reduced.

Q.No	Country	Article	Ref. in National Report
211	India	Article 19.4	Article 19.4.2 Page 177

Question/ Comment

It is stated – ‘Assuming failure of the vessel, the corium pours into the reactor pit. The strategy currently in place on the fleet in operation consists of injecting water (addition of water after vessel rupture or flooding of the reactor pit prior to vessel rupture).

As the safeguard systems of the damaged plant unit were probably lost on entry into the SA, the emergency team can implement ‘ultimate’ alignment to find the corium’.

What is the efficacy of corium cooling when water is injected on the top of corium? Does this strategy help in retarding/stopping molten corium concrete interaction?

Answer

As indicated in p.177, EDF has decided for its fleet, to use a generator-driven pump supplied by the Ultimate Emergency Diesel, allowing injection of water into the primary system. This system will belong to the hardened safety core and will reduce the risk of severe accident.

Assuming failure of the vessel, the corium would pour into the reactor cavity where it would be flooded by water. Several R&D programs in which EDF has been involved (MACE, CCI-OECD) have been performed to study molten core concrete interaction with flooding of the corium. These programs have concluded that various phenomena could increase the heat transfer between the corium and the water (corium ejection, water ingression, bulk cooling) which helps in delaying and, in some cases, stopping the MCCI. The efficiency of corium flooding depends on various parameters such as concrete composition and time of flooding. Studies are in progress at EDF to optimize this kind of solution to significantly reduce the risk of basemat melt-through in case of severe accident with failure of the vessel. In this context, the implementation of a spreading area in a room neighbouring the reactor cavity could be considered in some cases if cooling the corium in the cavity would not be sufficient.

Q.No	Country	Article	Ref. in National Report
212	Japan	Article 19.4	p181

Question/ Comment French report says that new baseline safety requirements concerning the on-site emergency plans improve EDF□fs preparedness for the management of emergency situation.

How are the long-term post-accident management, such as, operation and maintenance of components, required? Management may include on-site organization, inspection and repair of components and others.

Answer The accident studies included in the safety analysis report (SAR) consider a “long term” phase, starting after the first manual action. This part of the studies aim at demonstrating that, following the accident, a safe shutdown state of the can be reached using the safety systems plant. A safe shutdown state is a sate where the reactor core is sub-critical, the residual power is evacuated and the confinement is assured.

In a generic letter regarding the long term operation of existing reactors, ASN has requested EDF “to check that every new equipment (hardware and instrumentation), is qualified under severe accident conditions for the mission duration needed in case of an accident with core melting.”

Moreover, as a sequel of post-Fukushima complementary safety assessments, ASN has issued a set of decisions requiring the implementation of a “hardened safety core”. in particular, prescription ECS-ND9 provides that

“The new hardened safety core SSC which cannot be replaced by other means are the subject of reinforced design and manufacturing requirements to ensure a high level of reliability enabling them to perform their safety functions in all phases of an accident, for as long as they are needed.” This can be for example ensured through the support the emergency teams (normal emergency teams and/or nuclear rapid response force implemented after the accident at Fukushima Daiichi NPP).

Q.No	Country	Article	Ref. in National Report
213	Romania	Article 19.4	19.4.2

Question/ Comment On page 176, in the section describing the measures to cope with the slow pressurisation of the containment, it is mentioned that the containment venting-filtration system "opens after 24 hours as from a minimum pressure equal to the containment design pressure".

Please clarify whether the system is operated as soon as the pressure in the containment reaches the design pressure or at a different pressure.

Answer From an Operating Procedure point of view, opening of FCVS is only linked to the containment pressure.

The FCVS system shall never be opened before the containment design pressure (about 5 bar abs) is reached. This pressure is not reached before at least 1 day after the onset of the accident. This minimum period of 1 day before opening of the FCVS allows the Authorities to implement the adequate measures for protection of the population (sheltering, distribution of stable iodine, evacuation).

NPPs safety is fundamentally based on confinement of Fission Products inside the containment building. This is not modified by the

implementation of a FCVS. In consequence, opening of the FCVS has to be considered as an ultimate protection of the containment integrity against slow over pressurisation, postponed as long as possible, and if possible avoided.
According to this fundamental safety principle, National and Local Crisis Teams have the possibility, considering all the parameters they have in hands concerning the global management of the SA, to recommend to postpone FCVS opening above the containment design pressure. In this case the highest value possible, considering all the NPPs and all containment buildings resistance singularities (in particular hatches), is presently 6 bar abs.
In any case, the Plant Crisis Manager has the (final) responsibility of the opening of the FCVS after consultation of the National and Local Authorities involved in the SA crisis management.

Q.No	Country	Article	Ref. in National Report
214	Romania	Article 19.4	11.2.4

Question/ Comment: It was mentioned in Section 11.2.4 ASN analysis and oversight, on page 79, that "during the targeted inspections performed in 2011 as part of the Fukushima Daiichi accident experience feedback process, ASN in particular checked personnel training for severe accident management. These inspections confirmed that the training and qualification of the personnel in the facilities was on the whole satisfactory, even though a few deviations were observed on the documentation or monitoring of the training on some sites."

Has ASN implemented a periodic inspection programme to verify licensees' training for accident management, including severe accident management? What guidelines and criteria do ASN inspectors use to verify the licensees' training in accident management?

Answer: Licensees' training for accident management, including severe accident management, is regularly inspected as part of the standard inspection program that provides that such inspection should be performed every two to three years.
Specific internal guidance is available to the ASN inspectors.

Q.No	Country	Article	Ref. in National Report
215	Romania	Article 19.4	General

Question/ Comment: The report submitted by France is very comprehensive and detailed.

Answer: France really appreciates this comment.

Q.No	Country	Article	Ref. in National Report
216	Spain	Article 19.4	176

Question/ Comment: EDF has started to study the possibilities of improving the containment venting-filtration system.
What are the requirements and dates imposed by ANS on those systems after stress tests? Are they to be part of the hardened safety core? What are the seismic design specifications for existing and improved systems? Is there any result on the H2 risk study EDF is

	doing relating these systems yet?
Answer	EDF position has been submitted to ASN at the end of 2013. The technical solution proposed by EDF is to keep a closed containment even in case of SA including a core melt, for limiting as much as possible radiological releases to the environment. To achieve this, EDF has decided to implement, as part of the “hardened safety core"", new dedicated SSCs to remove the decay heat from the containment building with a recirculation loop cooled by a dedicated heat sink. In addition, in the frame of the “defence in depth” concept, to ensure an ultimate protection of the containment, and taking into account the Fukushima experience, seismic resistance of the existing FCVS installed on the French plants will be increased. As the FCVS doesn’t belong to the “hardened safety core"", the level of seismic resistance taken into account will be less than for the hardened safety core SSCs Moreover, in order to strongly limit gaseous iodine production, STB (Soda Tetra borate-Borax) baskets will be installed in the containment sumps of 1300 MWe and N4 NPPs which are not equipped only with Silver Indium-Cadmium control rods. Improvement of the filtration capacity of FCVS, particularly for organic iodine, has been studied but, according to the above decisions, the installation of such improved FCVS is not necessary. Concerning hydrogen combustion risk at the opening of FCVS: - Since the original design of FCVS, a preheating system is installed in order to avoid, by steam condensation, combustion of hydrogen in the system. Note: installation of PARs (Passive Autocatalytic Recombiners) in all French NPPs has very strongly decreased the occurrence of this potential risk, but, using deterministic considerations, it cannot be totally excluded in the case of accidental situations with late loss of core water injection leading to insufficient delay for sufficient hydrogen recombination before FCVS opening. - After recombination of all oxygen of the containment building by PARs, which could occur only in case of MCCI, there is a risk of combustion between hydrogen (and carbon monoxide) coming from the RB and oxygen initially present in the FCVS. This is a very transient risk. This situation is studied in the frame of the Plants Life Extension Program and more especially, in the field of the measures to ensure cooling of the core in the vessel or in the reactor pit which strongly will limit the production of combustible gases during MCCI and consequently will eliminate this kind of risk.

Q.No	Country	Article	Ref. in National Report
217	Spain	Article 19.4	176

Question/ Comment	Regarding SAMG, is there any plan to carry out a comprehensive reassessment and review in the light of the Fukushima lessons learnt?
Answer	The ability to manage a situation of severe accident is based on material and organizational arrangements (which includes SAMG). The Fukushima event led EDF to re-examine on these two levels. Following the analysis of the Fukushima event, no necessity to carry

out a comprehensive reassessment or review of our SAMG has been identified. However, some new equipments will be installed in each nuclear power plant to further avoid a severe accident (pumps, diesel, I&C, ...). When these equipments can be used during severe accident, they are taken into account in the updating of the SAMG according to the deployment of their integration in the power plant. Moreover, the update of the SAMG takes into account some intellectual updates such as the monitoring of the spent fuel pool. Nota : the lessons of Fukushima accident confirm the need for devices, already present in EDF NPPs and taken into account in the SAMG, such as the hydrogen recombiners and the containment decompression system in order to preserve containment integrity. In the light of the Fukushima accident, the ability to act in such kind of situation has been re-examined. Specific provisions are made in this field, with respect to the protection of the intervening teams on the plant : personal protective equipment, material modification to limit releases of radioactive iodine (implementation of tetra borate baskets in the containment sump), ... These provisions are not carried by the SAMG but reinforce the ability to implement the actions defined in this guide. Furthermore, the SAMG is tested during crisis exercises by using the full scope simulator with operating team to ensure the ability to implement the actions defined in this guide. In the light of Fukushima accident, EDF has also two grounds of research which are “rebuilding communications network on damaged site” and “how to acquire knowledge from the field to protect plant operators”.

Q.No	Country	Article	Ref. in National Report
218	Korea, Republic of	Article 19.6	185

Question/ Comment: Please explain how to calculate the number of significant safety events classified on the INES scale in the article 19.6.2, 0.9/plant unit per year in 2011 and 1.55/plant unit per year in 2012.

Answer: This number is calculated by dividing the number of events classified 1 or more on the INES scale on EDF NPPs in the year by the number of EDF NPPs. In 2011 and 2012, the total number for the 58 EDF NPP was 53 and 90.

Q.No	Country	Article	Ref. in National Report
219	United Arab Emirates	Article 19.6	184

Question/ Comment: The National Report states that ASN reviews EDF event notifications to determine whether further regulatory activities are needed such as inspections. Please describe the methodology and any criteria used in these analyses to determine the regulatory response to NPP notifications.

Answer: After an event is notified by EDF, ASN checks the analysis made by the licensee following the event, especially the impact of the event safety (real and potential), the lessons learned, the preventive, corrective and curative actions decided and their implementation schedule. After receiving the detailed analysis report of the event from EDF, ASN can also decide further investigation and, if needed, further regulatory actions.

The regulatory response by ASN is proportionate in the stakes, regarding risk prevention and inconveniences for the safety (security), the health and public health or the protection of the nature and the environment, mostly on a case-by-case analysis (frequency, deviation in the safety culture, generic issue), with the support from its TSO (IRSN).

	As necessary, ASN performs on site inspections after the notification, for instance in case of activation of the emergency plan, when the specific operating rules for managing an incident or an accident had to be applied, if any impact on the environment is suspected or in case of media impact. In case of a regulatory non-compliance, a specific process of enforcement can also be applied.		
Q.No 220	Country United Arab Emirates	Article Article 19.6	Ref. in National Report 185
Question/ Comment	ASN indicates that it has the power to initiate a technical inquiry in the event of an incident or accident. Please describe further the criteria that are considered to initiate an inquiry, describe how an inquiry may be different from a regulatory inspection, and identify any NPP inquiries in recent history.		
Answer	Regular regulatory inspections are organized each year. In case of an incident, an inspection will be organized on site rapidly, in the following days. The inspection will be based on the understanding of the event and the identification of the gap with the reference documents applicable. In an emergency situation, the ASN Commission could also impose actions to the licensee through ASN resolutions. In addition, the prosecutor will be in charge of the inquiry for severe situations.		
Q.No 221	Country United States of America	Article Article 19.6	Ref. in National Report 19.6.4
Question/ Comment	The report states that the number of events significant for the environment (ESEs) is down since last year but remains high compared with previous years. It also states that environmental protection must remain one of the core concerns of EDF. Please explain what actions the EDF is taking to reduce ESEs.		
Answer	The increase of the number of events significant for the environment (ESE) compared to the year 2007 is due to the introduction of a new criteria relative to the refrigerant gas emissions (as used in air-conditioner devices...) in the last quarter of 2007. Since this change of the event criteria, the first cause of ESE is this type of events : 60% of all the declarations, that is to say about 3 events per plant per year. The level of declaration is relatively low (20 kg) regarding the environmental impact. Nevertheless, EDF utilities keeps on working to maintain a low level of emissions and to reduce the number of events, working both on surveillance (containment control) and maintenance. The other causes are diverse : about 40 events on all the fleet. EDF Nuclear operating Division at the national level analyse continuously the different causes of these events (equipment, organisational or human), share with all the other plants the findings of the analysis in order to reinforce and prevent the occurrence of these events and if necessary, modify the operating procedures . All these actions have allowed the decrease of the number of events during the last years, confirmed in 2013.		
Q.No 222	Country Germany	Article Article 19.7	Ref. in National Report 19.7.4.1, p.191
Question/	[…] The available information concerning the manufacturing practices in force since the early 1970s in France gives no indication of		

Comment	the presence on the French NPP reactor vessels of manufacturing defects in numbers and of dimensions similar to those discovered on the Doel 3 reactor vessel. ASN has nonetheless asked EDF to carry out a detailed documentary review to confirm the correct performance of the manufacturing completion inspections. ASN also asked EDF to propose an inspection programme for certain vessels in order to further confirm the guarantees given. Did the documentary review already yield any results? Please give further details on the structure of the inspection program.
Answer	To provide complementary information from those obtained with the documentation resulting from the manufacturing guarantees , ASN asked EDF to conduct analytical work recorded on the occasion of each ten years outage. To date 40 pressure vessels were under review. EDF analyzed records on 12 of them and his work will be completed in the spring of 2014. At this stage EDF revealed some defects of small dimensions for some vessels, but no matching either in quantity or size to those observed in Belgium. The same examination is implemented systematically on vessels whose ten-years inspection is coming. In addition, ASN asked EDF to conduct a review of the entire thickness of the vessel on 6 reactors, which represents 10% of the pressure vessels. These tests are currently under progress and will be completed during the summer of 2014. In 2013, this review was conducted on the pressure vessels BUGEY 3, BLA 2 and DAM 3 and showed no defect. At this point, ASN considers that the probability of finding identical to those found on pressure vessels of Doel 3 and Tihange 2 is low.

Q.No	Country	Article	Ref. in National Report
223	Luxembourg	Article 19.7	page 178

Question/ Comment	What are the complementary measures mentioned on page 178 concerning the control of the reactor in case of habitability problems of the control room? In how far is the option of a secondary, physically separated or remote control room considered?
Answer	In the stress tests, it has been postulated that all the electrical supplies were lost. So, the control room ventilation and filtration system is not available. The leaks of the double wall containment are not filtrated either, and the pre-heating of the venting line of the containment is also unavailable, inducing a potential risk of H2 explosion. In such conditions operators couldn’t stay permanently in the control room during the period following the opening of the FCVS system. Following the stress tests, it was consequently decided to implement a small fixed diesel generator to supply ventilation and filtration of both the control room and of the double wall containment annulus, and also a mobile diesel generator brought by the FARN to supply the FCVS pre-heating. The “bunkered” ultimate emergency diesel generator (belonging to the hardened safety core) will also supply these systems when installed. Moreover, on four loops plant, baskets of soda tetra borate will be implemented to maintain basic the water in the sumps in order to limit organic iodine releases. Thanks to these modifications, the control room has not to be evacuated, even temporarily, after a severe accident. Eventually, EDF has decided to implement, as part of the “hardened safety core"", a new dedicated system to remove the decay heat

from the containment building in case of severe accident without venting of the containment. The radioactive releases and dosimetry on the site will be still more reduced.
Consequently, a secondary control room has not been considered.

Q.No	Country	Article	Ref. in National Report
224	Luxembourg	Article 19.7	page 176

Question/ Comment	About the containment venting systems, the report mentions on page 176 that this is opened after 24 hours. How does this work? Is there an automatic opening if certain conditions are met, or does it need manual opening? Once opened will it stay opened?
Answer	The opening of the FCVS is not automatic. The rational for this is described here after. NPPs safety is fundamentally based on confinement of Fission Products in the containment. This is not modified by the implementation of a FCVS. In consequence, opening of the FCVS has to be considered as an ultimate protection of the containment integrity against slow over pressurisation, postponed as long as possible and avoided if possible. Consequently: - Opening of the FCVS is a concerted action, considering the long grace period (at least 24 hours). - Early opening, due for example to transient hydrogen combustion in the containment building has to be avoided. This lead to avoid automatic actuation such as performed by a rupture disk and to use valves for opening the FCVS. By design, to take into account loss of power during SA, the two containment isolation valves are manual.

Q.No	Country	Article	Ref. in National Report
225	Korea, Republic of	Article 19.8	193

Question/ Comment	Is there any specific or quantitative criteria of ASN on "waste zoning" distinguishing the areas of the facilities with waste contaminated by radioactive material or activated by radiation from the areas with waste not containing radioactive material?
Answer	In a Nuclear Basic Installation (BNI), the methodology to decide whether a material is considered as radioactive relies on the waste zoning concept. The waste zoning consists in distinguishing zones of the facility where the waste is likely to have been contaminated with radioactive substances or activated by radiation (zones called “nuclear waste zones”), and zones where the waste is not likely to be contaminated or activated (zones called “conventional zones”). This concept was originally set up by the decree of 31 December 1999, now being replaced by the ministerial order of 7 February 2012 (taking effect from 1st July 2013). Details are provided in an ASN guide. A “zone” is a room, part of a room, or part of an installation for which boundaries or physical barriers exists and can be deemed to prevent any transfer of contamination between the outside and the inside of the zone. Thus the possible interruptions of the physical barriers must be considered very carefully. The licensee determines the waste zoning on the following bases: design of the installation, operational procedures, history of the installation (incidents, modification, controls, etc.).It is reminded that this approach constitutes the first line of defence, the others being radiological controls of the waste considered as conventional according to the waste zoning. The licensee has to submit a waste survey to ASN for approval. Of course this document includes the definition and justification of the

proposed waste zones.
In addition, inspections are conducted on site by ASN on this subject. In conclusion, there is no quantitative criteria in order to define "waste zoning" but only a qualitative one.

Q.No	Country	Article	Ref. in National Report
226	Korea, Republic of	Article 19.8	2, 195

Question/ Comment: It is stated that "the administrative shutdown of the CENTRACO incineration unit following the accident that occurred on the melting unit on 12th September 2011".
What are the cause, result and lessons learned from the accident?

Answer: Centraco facility is mainly composed of two installations: an incinerator (3.500 t/yr of solid and liquid LLW) and an induction furnace (melting of 1.500 t/yr of scraps slightly contaminated). On September 12, 2011 an accident (explosion) occurred at the work level of the furnace, followed by a fire which has been brought under control within one hour while the melting process was shut down. A worker died, burned by the melted metal ejection, and four others were injured (one seriously). At the present time, two of three inquiries launched after the accident are still in progress. Seals are always affixed to the melting unit of Centraco facility. It is premature to evocate the causes of the accident. Following this event, it was also decided to stop the incineration process
(still stopped 6 months later - March 2012). More than 80 % (in mass) of DAW waste (paper, plastics, clothes...conditioned in plastic drums) arising from operation of EDF PWR fleet
used to be incinerated. It was decided that NPPs would commit in the sorting of DAW because 70 % of them are accepted into the repository Centre of Aube, when pre-compacted
in metallic drums. This option has contributed to limit the quantity of DAW in interim storage on sites waiting for the commissioning of the incinerator.
Concerning liquid waste, all 1300/1450 MWe reactors' evaporator bottom concentrates, enriched in boron, used also to be incinerated. EDF restarted a mobile machine able to manage these concentrates by cementation directly in concrete containers.

Q.No	Country	Article	Ref. in National Report
227	Korea, Republic of	Article 19.8	4.2, 196

Question/ Comment: What are the specific requirements to be issued from ASN in particular to reinforce the electrical power resource, the water supply resource, the instrumentation and the measures to prevent accidental emptying of the pools in relation to the stress test of EDF?

Answer: ASN required EDF to complete the pool instrumentation with electrically secured for accidental conditions water level and temperature sensors. Within the hardened safety core a secured alternative water supply will be implemented using an independent water source.
In addition:
1. EDF extended the diameter of passive devices to avoid siphoning. These devices are designed in order to avoid the leakage of the

water by the pool after an eventual pipe break of fuel cooling system
2. EDF decided to realize more of inspections in the spent fuel pool
3. EDF installed a waterproofing joint in the two sides of the swimming pool reactor in order to avoid accidental emptying of the pools
4. EDF installed an automatic shut off of suction side relevant spent fuel pool cooling system
5. EDF motorized the transfer pipe valve and the control is managed from the control room
6. EDF is checking the structural integrity of the pool under beyond design earthquake and heavy load drops.

Q.No	Country	Article	Ref. in National Report
228	Korea, Republic of	Article 19.8	1.2, 194

Question/ Comment	What is the management plan for spent fuels under abnormal conditions in light of the Fukushima Daiichi accident?
Answer	The strategy for the spent fuel under abnormal conditions is to guarantee by the implementation of systems structures and components that the fuel will remain under water under all circumstances. This point is achieved by the definition of beyond design conditions and the verification of the pool capacity to retain and be fulfilled with adequate amount of water with additional secured and redundant means (water source, piping, electrical systems, I&C…). The global strategy is to check the pool integrity, including non isolable connected circuits under beyond design external hazard (20.000 years return period earthquake) and internal hazards (heavy loads drop) and to implement reinforced water feeding systems with a redundant system, electrically secured with an alternate water source. In case of failure of these SSC’s, the possible implementation of mobile means remains possible.