# Convention on Nuclear Safety

## 3rd Review Meeting

## Vienna - 11-22 April 2005

# France's answers

### to questions and comments received from other Contracting Parties on its 3rd report for the CNS

**1st April 2005**

Foreword

*Questions and their answers are listed along the Articles of the Convention and, within each article, sorted by arrival order.*

*When a similar question has been raised several times, the answer is repeated for reading facilitation within the addition of the mention of the question number where it firstly occurred.*

*When a question includes several sub-questions, generally indicated by Q1, Q2, etc, the answers to the various sub-questions have been in the same way preceded by the mention Q1, Q2, etc.*

**Convention on Nuclear Safety**
Questions Posted To France in 2005

**General**

| Seq. No 1 | Country | Article General | Ref. in National Report |
|---|---|---|---|

Question/ Comment: Our Country compliments France for voluntarily including information abut research reactor in its National Report and notes that the regulations for research reactors are generally the same as those for power reactors, in relation to radiation protection and nuclear safety, and that the Code of conduct for Research Reactors incorporated most of the provisions from the Nuclear Safety Convention.

Answer: France is thankful for this comment.

Indeed, at the 1st CNS Review Meeting (1999), France did not report about the power reactor Phenix since it is both a power reactor and a research reactor and it was one of the main criticism provided to France by the Review Meeting Therefore France added the Phenix reactor to its report for the 2nd CNS Review Meeting and it was well received.
Based on this experience as well as on the reporting about all research reactors due under the Joint Convention to which France is party, and since there is no difference in French regulation between power reactors and research reactors, France has decided to extend its CNS reporting for the 3rd CNS Review Meeting to all French Research Reactors.

| Seq. No 2 | Country | Article General | Ref. in National Report p. 5 + § 6.4 p. 20 |
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Question/ Comment: Although not obliged to do so under the terms of the Convention, the French authorities are to be commended in including research reactors in the National Report.

Answer: France is thankful for this comment

Indeed, at the 1st CNS Review Meeting (1999), France did not report about the power reactor Phenix since it is both a power reactor and a research reactor and it was one of the main criticism provided to France by the Review Meeting Therefore France added the Phenix reactor to its report for the 2nd CNS Review Meeting and it was well received.
Based on this experience as well as on the reporting about all research reactors due under the Joint Convention to which France is party, and since there is no difference in French regulation between power reactors and research reactors, France has decided to extend its CNS reporting for the 3rd CNS Review Meeting to all French Research Reactors.

| Seq. No 3 | Country | Article General | Ref. in National Report |
|---|---|---|---|

Question/ Comment: General comments to 2. Main changes with respect to the 2nd French report
Many good practices in order to enhance nuclear safety were found in the French national report. Especially the following aspects should be noticed and appreciated.
- Establishing a safety management inspections system and annual meeting of the

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| | Advisory committee (2nd para. of 2.3.4, page 9).<br>- In order to verify the nuclear safety under the extreme weather related problems, EDF will reassess the effects of external events on a NPP, one after another, within the framework of the regular outages of NPPs. (paras.2 and 3 of 2.3.6, page 10)<br>- 8.1.4 on page 42 and 4th and 5th paras. of 8.1.5.1 on page 43;<br>-- the ASN has been following a policy of service company diversification, both nationally and internationally<br>-- Each Committee can call on anyone whose competence they feel would be useful. … The participation of foreign experts can lead to further and more diverse approaches to problems and offer greater benefit from experience acquired internationally.<br>- 10.2 Measures taken for power reactors; EDF is continuously improving and enhancing nuclear safety putting on the utmost importance to safety. Clearly-defined safety responsibilities should be the bases of safety improvement and responsibility of all players should reinforce nuclear safety in France. It should be noted that EDF created an inspection and verification system in each division of the Nuclear Operations Department.<br>- Human factors are also taken into account in subcontracting and in monitoring of service companies. In particular, a “service company safety quality” training course has been setup…(The last para. of 12.3.1, page 58)<br>- Development of emergency preparedness and its practice involving the neighboring countries (16 Emergency preparedness) |
| Answer | France is thankful for this comment. |

| Seq. No 4 | Country | Article General | Ref. in National Report |
|---|---|---|---|
| Question/ Comment | The report is well structured, clear and discusses all relevant aspects of nuclear and radiation safety from both the regulatory and the operators' side in depth. | | |
| Answer | France is thankful for this comment | | |

| Seq. No 5 | Country | Article General | Ref. in National Report |
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| Question/ Comment | The report, as expected, is exceptionally well written. Its introductory chapters are useful, and the idea of voluntarily including a 20th concluding chapter summarising planned activities to improve safety is one which the UK should consider copying for future reports. There are few implied criticisms in the questions –most simply seek further information. | | |
| Answer | France is thankful for this comment | | |

**Art. 6 – Existing nuclear installations**

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 6 | | Article 6 | Para 6.4.2 |

Question/ Comment: For what period of time is a licence to operate a research reactor valid?

Answer: In the French legislation there is no time limit in the operating licence of a Basic Nuclear Installation (including power reactors and research reactors), provided the Nuclear Installation continues to fulfil at any time the safety requirements set by the regulatory authority (which may imply backfitting following Periodic Safety Reviews). Depending on the conclusions of the Periodic Safety Review, in order to ensure medium or long term satisfactory safety conditions, an upgrade of the Nuclear Installation or compensatory provisions may be necessary if the operator does not plan to stop operating the Nuclear Installation.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 7 | | Article 6 | Para 6.4.2 circa |

Question/ Comment: Does ASN have a requirement for periodic review and revision by the operating organisation and review by the ASN of the safety case for research reactors?

Answer: As indicated in the report, research reactors belong to the category of "Basic Nuclear Installation", which are regulated by decree 63-1228 of 11 December 1963. This decree, by its article 5 (see report p. 68) enables the ministers in charge of nuclear safety to request the operator at any time to conduct a periodic safety review of an installation. In practice the Nuclear safety authority (ASN) informed the operators that a periodic safety review was necessary about every ten years.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 8 | | Article 6 | § 2.3.6 + p. 16-20 |

Question/ Comment: It is noted that following clogging of water intakes at two plants in 2003, EDF shut them down as a precautionary measure. Was there at any time a concern that the availability of cooling water would be insufficient to ensure cooling in the shut-down mode?

Answer: In December 2003, extraordinary bad weather caused the Rhône to rise quickly with a considerable inflow of mud and vegetable waste at the inlet of the pumping stations at Cruas (4 900 MW plant units) and Tricastin (4 900 MW plant units). The arrival of this fouling matter caused faster clogging of the Component Cooling System exchangers requiring preventive fall back on the Cruas plant units 3 and 4 and of the Tricastin plant unit 4, which were the most exposed plant units considering the configuration of the pumping stations.
Automatic shutdown of the Tricastin plant unit 3 actually took place consecutively to the trip of the circulating pumps on the condenser.

Although no safety criteria was challenged, the on-site emergency plan was triggered as a preventive measure, to allow more effective management of the event, at local and corporate level.
Preventive fallback of part of the plant units and the alternating cleaning of the Component Cooling System exchangers made possible to avoid at any time the unavailability of the heat sink for all the eight reactor units at Cruas and Tricastin.

| Seq. No<br>9 | Country | Article<br>Article 6 | Ref. in National Report<br>§ 6.3.2 - p . 17 |
|---|---|---|---|

Question/ Comment: In relation to flooding, what is the basis for the safety case that has resulted in ASN allowing until 2008 for the completion of waterproofing underground structures? Also which nuclear power plants will be subject to this upgrading?

Answer: Following the flooding of Le Blayais, EDF decided to implement a "volumetric protection" on all the French NPPs, in addition to the conventional protection (dams, walls, etc.). Thus all NPPs will more or less be the subject of this upgrading.

The principle of this "volumetric protection" consists in protecting the infrastructure of the buildings containing safety related equipment, through waterproofing underground structures, necessary to attain or maintain a safe state in the event of external flooding. EDF presented to ASN the technical challenges related to the realization of these volumetric protections. Those require in particular the design of materials, which do not exist currently on the market. Considering these problems added to the amount of work on all the sites, the ASN considers 2008 to be a realistic and acceptable deadline for the completion of work.

| Seq. No<br>10 | Country | Article<br>Article 6 | Ref. in National Report<br>Paragraph 6.3.7 |
|---|---|---|---|

Question/ Comment: It is noted that the dossier submitted for approval by ASN includes, inter alia, a reference to the management of any leaks of water from the pool, even if it has been shown to be properly watertight. What, in relation to this statement, does the term “management” encompass?

Answer: The term "management' of the leaks mentioned in the statement of § 6.3.7 of the report refers to the following items:

- The maximum leaks considered are low enough to be compensated for by input without any risks of cooling being lost.
- The maximum envisioned leakage is compatible with the capacity of the drainage network that exists under the sealing liner.
- The maximum leakage considered is low enough to be recovered by the Fuel building sump pumps without any risk of flooding the premises.

| Seq. No<br>11 | Country | Article<br>Article 6 | Ref. in National Report<br>Paragraph 6.4 |
|---|---|---|---|

Question/ Comment: In relation to the Phenix reactor, which first went critical in August 1973, has a projected closure date been established? In relation to other research reactors in France, how many of these are operational?

Answer: In December 1997, the French Nuclear Safety Authority has requested that the CEA carries out various operations to check out the Phenix reactor state and upgrade it in conformity with new safety regulations. These operations are summarized in the second and the third reports under this Nuclear Safety Convention.
In 2003, at the end of these upgrade operations, Phenix was authorized by the Nuclear Safety Authority to be operated at two third of the full power for six cycles

of 120 Equivalent Full Power Days (180 days at two third of full power). The second cycle is currently under way. The objective of these six cycles is to carry irradiation experiments concerning transmutation and future fuel materials.

For what concerns the French research reactors, Phenix included, the 10 presented in the report are in operation: 5 in Cadarache, 1 in Grenoble, 1 in Marcoule, 3 in Saclay.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 12 | | Article 6 | p.16 |

Question/ Comment

Reference : 2.3.8,p11 and 6.3.1,p16

Q1: You report that PSA is performed in your country (2.3.8,p11 and 6.3.1,p16). We would like to know the outline of the execution plan of the risk informed in-service inspection (RIISI) utilizing the risk information deducted from PSA. Is your government going to permit the RIISI according to the plant application? Or request the RIISI to all nuclear power plants?

Q2: When you establish codes and standards for the RIISI in France, are you going to refer codes and standards of United States? Or are you going to develop your original codes and standards?

Answer

Q1:

The ASN does not use the risk informed in-service inspection (RIISI) for passive components but uses a deterministic approach.

However, the ASN is currently examining the maintenance method for active, mainly electromechanical components (pumps, valves, etc.) set up by EDF in the mid-90s. This method, which was based on American practices known as “Reliability Centred Maintenance”, was adapted by EDF under the name of “optimisation de la maintenance par la fiabilité” (OMF). It aims at improving the efficiency, rationality and traceability of the basic preventive maintenance programmes, in terms of safety, availability and cost issues. The OMF method uses a functional approach, which determines what maintenance has to be performed according to the consequences of equipment failure, rather than simply according to its causes, as it is the case under a conventional approach.

Q2:

The development of codes and standards refers to the regulation as established by the ASN and is of the responsibility of the operators. Thus, French regulatory practice with respect to nuclear safety requires the plant operator to submit a document defining the rules, codes and standards he will implement for the design, construction, start-up and operation of safety-related equipment. The codes provide a means of both complying with general technical regulations and upholding good industrial practice. This gave rise to formulation by the French manufacturers of French design and construction rules, known as the RCC codes which, for the different categories of equipment involved (civil engineering, mechanical and electrical equipment, fuel, etc.) concern the design, construction and operation stages. These documents are drawn up by the manufacturers, which may take benefit of international experience, and not by the Nuclear Safety Authority, which nevertheless examines them in detail, both in their initial and revised versions. In most cases, their contents are then integrated into a Basic Safety Rule, thereby confirming their relevance at the time of publication.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 13 | | Article 6 | p. 16-20 |

Question/ Comment: The existing situation in the countries with nuclear power programs is characterized by the need for more frequent upgrading of control systems as compared to NPP major process equipment since the lifetimes of automation features and process equipment differ by the factor of 3-5. Besides, fast progressing development of automation features does not allow to perform adequate replacement of the obsolete automatic controls with new, up-to-date ones. The appearance of programmable automation features with new capabilities to perform information and control functions is currently not quite properly substantiated in terms of reliable functioning, and this is noted in the IAEA and IEC documents. In this situation it is essential to have a well-reasoned concept of control systems upgrading that could be performed with no breach of NPP safe operation standards and regulations.
Does your country have a concept of upgrading safety-related control systems for all operating nuclear plants?

Answer: In France the concept of upgrading safety-related control systems is conditioned by the following criteria :

- the need of functional improvements which couldn’t be implemented in the technology initially used,
- the increase of the cost of maintenance or the decrease of the availability due to the numbers of failure,
- the obsolescence of the equipment and the lack of spare part.

Usually the safety-related control systems are kept as long as possible without modification or upgrade. This is possible thanks to an internal organization in charge of :

- the survey of the aging of the equipment,
- the definition of specific contract with some suppliers to warranty the everlastingness of the equipment,
- the survey of the industrial market.

In case a safety-related control system needs to be upgraded, the man-machine-interface is kept as similar as possible in the main control room in order not to be obliged to change the operating procedures.

Concerning the software implementation, each software modification as well as each new software must be in conformity with the relevant Basic Safety Rules. These ASN rules (RFS II.4.1.a) stipulate for example that for the highest level of safety (1E), the software must be deterministic.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 14 | | Article 6 | p. 16-20 |

Question/ Comment: Widespread use of programmable automation means to substitute human action at NPPs eventually results in a situation where these means are being offered and applied to implement safety-related functions, in particular, reactor emergency

protections. As is known, reliability of programmable automation means cannot be estimated quantitatively, while the qualitative justification can always be admitted as incomplete, which is noted in the IAEA and IEC documents. In this connection a question arises as to the need for justifying/demonstrating the applicability of programmable automation means for these purposes as well as availability of positive experience with their use.
Do you have good experience with justifying the applicability of digital programmable safety-related NPP protection systems implemented on recommendations from IAEA and IEC at operating nuclear plants?

Answer In France, EDF have twenty 1300 MWe units (started 20 years ago) and four 1450 MWe units (started 10 years ago) using a reactor protection system based on digital and programmable technology. In both cases these two systems are compliant with IEC 60880 “Software for computers in the safety systems of nuclear power plans”. We can say that we have a good experience with justifying the applicability of digital programmable safety-related NPP protection systems implemented on recommendations from IEC.

The Basic Safety Rule RFS II.4.1.a on the safety-related softwares details the requirements and justifications expected for the safety demonstration of software; for example, it requires that the most important systems for safety (class 1E) be deterministic.

| Seq. No | Country | Article | Ref. in National Report |
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| 15 | | Article 6 | p. 16-20 |

Question/ Comment 1) Based on the NPP lifetime extension concept, what is specifically being done to justify the possibility of NPP life extension?
2) Which plants are planned to be "life extended" and for how many years?

Answer Q 1/
The concept of "life extension" is not strictly applicable in France, insofar as the authorizations are given without mention of limit of lifespan. However, every ten years, at the end of the periodic safety review, ASN must come to a conclusion about the installation operation continuation until the next review. For this reason, the thirty years lifespan stage will constitute from the ageing point of view, an important milestone.

From this point of view, ASN requires of utility EDF to bring, for each reactor, the demonstration of its possible operation beyond the third ten-yearly outage, under conditions of satisfactory safety. The approach selected relies on the 3 following phases :

- preparation actions preliminary to the third ten-yearly outage;
- establishment of an aptitude file for each reactor for the operation continuation ;
- establishment, for each reactor, of a detailed management plan of ageing beyond the third ten-yearly outage.

On this basis, ASN will give an opinion at the end of the third ten-yearly outage, on the aptitude for reactors operation continuation.

The French policy for "plant life extension" after ten years is therefore related to a method of ten-year safety re-examination with in addition, and as a continuation, the examination of specific points with the Nuclear Safety Authority. These re-examinations are aimed at improving the safety level and relate to the performance of tests, modification (material, documentation, etc) and to specific normal and/or exceptional maintenance operations.

At present, the safety re-examination intended to allow the operation of 900 MWe plant units for the 30 to 40 years reference period is underway.
It is based on two main approaches:

- A conformity examination based on experience feedback from the previous conformity examinations of 900 MWe and 1300 MWe plant units and feedback from the 900 MWe units since the previous 10-year re-examination.
- A re-evaluation of the safety reference material fitting into the 30-40 years objective, consistent with the safety reference material derived from re-examination after 20 years for the 1300 MWe units, seeking to identify the themes to be privileged and the most pertinent evolutions possible.

As a complement, there is a provision for a Complementary Inspection Program which, in addition to the checks and inspections provided for in the maintenance documents, is aimed at seeking out any degradations not covered by these operating documents, concerning components or structures.

During this third 10-year inspection, the Safely Authority has also asked EDF to demonstrate its ability to control the aging of the components and structures for the 30-40 year period.
This demonstration of an aptitude to control the aging process will be carried out to the following methodology:

- Determination of important components and structures concerned for safety and the aging mechanisms to which they may be exposed.
- Demonstration of the correct control of aging of each "component or structure / aging mechanism" couple (480 couples processed).
- For the "component or structure / aging mechanism" couples for which aging control has not been demonstrated, the establishing of a Continued Operation Aptitude File presenting the component, the aging mechanism(s) and their consequences on the component or structure, and proposing a program of actions (R&D actions, studies, checks or inspections, modification of operating parameters, exceptional maintenance (repairs, replacements, etc) designed to ensure the control of aging for the period of 30 - 40 years (12 files dealing with 28 aging mechanisms for the 12 components concerned).This demonstration is completed by the steps taken by the company to guarantee the maintenance of operating skills of maintenance, engineering not only within the company itself but also among industrial partners and suppliers, and guarantee the durability of the supply of equipment or maintenance services.

In addition to the actions performed during the third safety re-examination, others are undertaken with a longer term perspective, that of 40 - 60 years but on main components such as the vessel, the pressurizer, the confinement enclosure, …to allow a technical-economic evaluation of the life duration capability of each 900 MWe plant unit after 40 years.

Q 2/
All the French 900 MWe reactor units are concerned by the actions underway.

During a second step, the 1300MWe reactor units then the 1400 MWe reactor units will be processed in the same manner.

| Seq. No 16 | Country | Article Article 6 | Ref. in National Report p. 16-20 |
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Question/ Comment
1) Do you take NPP power uprating efforts to increase reactor power above the design-specified level?
2) If so, at which plants did you complete these efforts and what are the results?

Answer
Q1/
There are several means to use the safety margins of the reactors. One possibility is to increase the reactor power. Up to now, EDF preferred to use these margins in order to increase the burnup rate, and to use the NPPs for the load follow. This appears more interesting for EDF due to the important part of the production made with NPPs (more than 80 %).
However, the feasibility of power increase has been demonstrated and tested on one plant (Chinon).

Q2/
To date no power increase has been authorised.

| Seq. No 17 | Country | Article Article 6 | Ref. in National Report § 2.3.2 + p. 16-20 |
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Question/ Comment
What corrective actions are being implemented to preclude damage to fuel rod cladding due to fuel rod vibrations caused by coolant flow in the French reactors of 1300 MW series?

Answer
The corrective actions have been implemented following a 4 steps process:

(1) An extensive analysis and interpretation of the observed phenomena in the 1300Mwe NPPs (wear mapping, flow measurements, thermal-hydraulic 3D calculations, manufacturing investigations, etc...) has been carried on,

(2) The main fretting wear root causes have been identified: extended operation time under conditions including reduced grid-to-rod support in a limited number of grid cells (due to End of Life (EOL) spring relaxation),

(3) The experimental qualification protocol to scale the robustness of a new fuel assembly design has been improved based on (1) and (2): a full size mock-up is tested for 1000h in an out-of-pile loop under different levels of cross-flow and different rod-to-grid support conditions (beginning and end-of-life conditions). The test results have been used to validate the calculation models,

(4) Design modifications to improve the grid-to-rod support of the lower part of the fuel assemblies have been proposed: a second grid has been added next to the lower grid in the Framatome design, a P-Grid has been added next to the bottom nozzle in the Westinghouse design. So far the in-reactor feedback of these improved designs has been very good.

| Seq. No 18 | Country | Article Article 6 | Ref. in National Report § 2.3.3 + p. 16-20 |
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Question/ In item 2.3.3 the loss of fuel rod leak-tightness due to wear caused by friction-

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| Comment | induced corrosion is treated as a safety problem.<br>1) Did the loss of fuel rod tightness at French PWRs of 1300 MW series lead to the breach of operation limits for fuel rod damage and coolant activity?<br>2) If the breach of operation limits actually occurred, how did you deal with the problem of continuing reactor operation with faulty fuel rods and high coolant activity? |
| Answer | Q1/<br>The operation of the reactor with a loss of tightness at the fuel rod is not a breach of the operating limits. The presence of a rod that is leaky in the reactor, and therefore of radiochemical activity in the primary circuit, is acceptable in operation within the limits of the operating technical specifications relative to the radio-chemistry of the coolant. These technical specifications define the operating authorization thresholds of the reactor for the concentration levels of some of the fission product isotopes in the primary water such as iodine 131, iodine 134 and xenon 133. Then they are reduced temporarily for the plant units concerned, pending the implementation of the workaround measures outlined in the report.<br><br>Q2/<br>The radiochemical specifications were complied with in spite of the rupture of more than 80 fuel rods at Cattenom 3 and many problems of radiation protection at the time following the unit outage. These consequences led ASN to lower significantly the thresholds mentioned in the radiochemical specifications so as to prevent the appearance of a new occurrence of major fretting corrosion. Currently, EDF is forbidden to reload non-tight fuel assemblies. |

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 19 | | Article 6 | § 2.3.3 + p. 16-20 |

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| Question/ Comment | In item 2.3.3 the risk of containment sump filters clogging with potential for ECCS and containment spray systems failure to perform their safety functions during large-break LOCAs is treated as a safety concern. Reassessment of the filters' design characteristics has resulted in a situation where the regulatory body requested EDF to state its position on the matter of filter clogging. Despite a large filtration surface area of the existing filters, EDF has admitted the need for resolving this problem for 900 MW PWRs and has planned the upgrading programs.<br>1) What specific engineering solution has EDF adopted to resolve the problem of containment sump filter clogging?<br>2) Can this solution assure the removal of smallest particles and thermal insulation fibre to avoid this kind of debris entry into ECCS heat exchangers?<br>3) Has this solution been approved by the regulatory body?<br>4) Did you explore the possibility of thermal insulation replacement? |
| Answer | Q1):<br>In order to solve the problem of sumps clogging, EDF has launched a test program on the proposed industrial solutions obtained via a call for bid. The purpose of theses tests is to precisely quantify the ability of these industrial solutions to achieve their function with sufficient margins.<br>These tests are based on a debris source term evaluated either on an international consensus (ZOI (Zone of Influence) : 12D for insulation material), on a French test |

program related to the coatings, and on multiple walkdowns in our plants which purposes are to determine the overall amount of particulates (dusts and others...) . All these debris are combined in order to take into account the worst case effect (e.g. : thin bed effect). The industrial solutions chosen are all based on passive means (sump grids) and the overall surface have been multiplied by an average factor of 15 (as a minimum).

Q2):
Debris can effectively enter the ECCS heat exchangers and the core via the pumps and return to the sumps where they will be trapped thanks to the formation of a thin bed of debris. This risk has been taken into account by qualification tests of the pumps with such debris, and by examining all the restricted passages downstream the sumps which led to the modification of certain taps of 900 MWe reactors.

Q3):
The reference documents, which have been used for the definition of the technical solution, have been examined by the safety Advisory Committee of experts for Nuclear Reactors and accepted by the Nuclear Safety Authority. However, additional justifications are requested which are under progress (e.g.: chemical effects and downstream effects). They will be globally released along the year 2005.

Q4):
EDF explored the possibility of replacement of the insulation type. There are several reasons, which lead EDF to withdraw as far as possible microtherm insulation where reasonable. But no major changes are foreseen since the new sumps can assume the present insulation. At the end, EDF concluded that the economical aspect of replacement is so high that it is far more interesting to favour new sumps solutions.

| Seq. No<br>20 | Country | Article<br>Article 6 | Ref. in National Report<br>§ 2.3.6 + p. 16-20 |
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Question/ Comment: What corrective actions are being taken to prevent the fouling/clogging of the cooling service water intake facilities at the plants susceptible to flooding in case of water level rise in the rivers, from which the service water is taken?

Answer: The risk of fouling/clogging, in flooding situation is taken into account in the design, by the trash rack systems and redundant filters and exchangers.
In case of an abnormal increase of turbidity or incoming debris, several defence lines can be successively used.

- First of all, filters and their washing system are run faster by anticipation.
- Then the flow taken in the river is decreased by stopping the non-safety related pumps, in order to reduce the quantity of incoming debris and the head loss through the filters.
- At last, in case of a safety train loss and during its repair, the cooling of the safety systems is performed by the second train.

In the hypothetical situation in which both safety trains would be temporarily out of service, the residual nuclear power of the reactor would then be evacuated by the steam generators. These SG are fed by demineralized water tanks, during the recovery time of filters and exchangers.

| Seq. No 21 | Country | Article Article 6 | Ref. in National Report Item 6.3.1 |
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Question/ Comment
1) Are all NPP reactor containments equipped with filters to cope with severe accidents?
2) Do you have regulatory requirements for equipping reactor containments with such filters or any criteria, which define the need for installing such devices?

Answer
Q1):
All French NPP are now equipped with sand bed filters to reduce risk of containment failure by over pressurization without releasing to the outside the amount of fission products which would be present in the containment in case of core melt.

Q2):
The implementation of these filters, which was not envisaged with the design for the first reactors, was decided by EDF in 1986 on all the reactors, within the framework of the insights after TMI. However, there is no legally binding requirement on this point in the French regulation. Since 1986, the technical objectives relative to these filters have been discussed with ASN. These prescribed objectives of containment filtered venting systems are:

- to limit pressure to a value between design pressure and "ultimate" failure pressure;
- to limit atmospheric releases to a value lower than the reference source term, compatible with measures defined by public authorities for protection of human populations (Minimum filtration efficiency: 10 for aerosols).

The main characteristics are:

- Time to opening: at least 24 hours (time to implement measures to protect human populations);
- No use of electrical power sources;
- Provision for measuring activity releases in inert gases, iodine and caesium;
- Installation in the Reactor Building of a metal mesh prefilter (which also leads to an increase in filtration efficiency for aerosols) to reduce potential site dose rate ("sky shine");
- Provision for preheating before opening to reduce Transient hydrogen risk on opening.

| Seq. No 22 | Country | Article Article 6 | Ref. in National Report |
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Question/ Comment
Some sections of the National Report (e.g. 6.4.1, 14.1.3.2, 14.4.1.1) mention the so-called notion of "safety level".
1) What is meant under the "safety level" and is there a quantitative measure of the "safety level"?
2) Is there a possibility to compare nuclear plants by their appropriate "safety levels"?

Answer
Q1):

The safety level cannot be easily quantified and is more an appreciation of the safety of French BNI based on the consideration of the design, the operation, the deterministic and probabilistic safety demonstrations.
Q2):
When performing periodic safety reviews, ASN reconsiders the original safety demonstration of the BNI and compare its safety level with the one of the most recent nuclear installations.

| Seq. No 23 | Country | Article Article 6 | Ref. in National Report section 6.3.3, p 17 |
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Question/ Comment This section mentions seismic issues. Certain spectra reveal that for the high frequencies some values can be higher than in the reactor design basis spectra used. These spectra will be used for seismic reassessment as part of the third 10-yearly outage programme for the 900 MWe and during the second 10-yearly outage programme for the 1300 MWe reactors.
1/ What would be the ASN response in case seismic reassessment will results in reduced safety margins with regard to the design basis?
2/ Can the PSA (probabilistic safety assessment) approach justify the reduced safety margins?

Answer Q1/
It is on the basis of the spectra defined in the Basic Safety Rule RFS 2001-01 that the seismic re-evaluation of the second 10-yearly outage programme for 1300 MWe reactors and the third 10-yearly outage programme for 900 MWe reactors have been undertaken. The basic approach consists of assuming that seismic conditions similar to historically known earthquakes are likely to occur in the future with an epicentre position which is the most penalising with regard to its effects in terms of MSK intensity on the site being compatible with geological and seismic data.
The safe shutdown, the fuel residual heat removal and the containment of radioactive products have to be assured for such earthquakes.
When the level of safe shutdown earthquake is superior to that of the design basis earthquake, notably at high frequencies, a verification process for the civil engineering structure and materials is undertaken on the concerned sites. At the end of this process, if the stability of the building cannot be demonstrated for the safe shutdown earthquake, reinforcement has to be implemented, if so decided by ASN.

Q2/
PSAs are not used to support such decision yet, since they do not encompass external hazards in France.

| Seq. No 24 | Country | Article Article 6 | Ref. in National Report 6.3.3 p.17 |
|---|---|---|---|

Question/ Comment The National Report refers to “ground motion response spectra”. Does DGSNR require the licensee to use seismic PSA, in which seismic events of different magnitudes, together with their associated frequencies, are used as initiating events in the overall PSA for the Basic Nuclear Installation (BNI), and are thus fully integrated into the calculation of risk from the BNI?

Answer At the moment, EDF has not developed a seismic PSA yet. Nevertheless, the

seismic risk for French BNIs is taken into account on a deterministic viewpoint. The basic approach consists of assuming that seismic conditions similar to historically known earthquakes are likely to occur in the future with an epicentre position which is the most penalising with regard to its effects in terms of MSK intensity on the site being compatible with geological and seismic data.
The safe shutdown, the residual heat removal and the containment of radioactive products have to be assured for such earthquakes.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 25 | | Article 6 | 6.3.4 p.18 |

Question/ Comment: Why are there no regulations applicable to BNIs which address the problem of Legionnaire's disease?

Answer: The question of risks related to the development of Legionnaire's bacteria in the cooling towers of the French nuclear power plants (NPP) is identified for several years by EDF, which leads in particular the operator to perform epidemiological studies. These studies have been presented to the ASN and to the public organisations in charge of health. One outcome of these studies is that no Legionnaire's bacteria cases were observed up to now close from a NPP. As one characteristic of the towers is their high size as compared to other industrial towers (around 150 meters in most of cases, except the Chinon B plant with a size around 30 meters), a consequence is the height of the release that is a factor, which contributes to reduce the risks.

Considering these elements, it has been judged that it was not necessary to impose limits similar to those imposed to other industrial installations, but that it would be fruitful forward to get values adapted to the nuclear plants.

Following the Legionnaire's disease epidemic, which occurred at the end of 2003 in the north region of France, the Public Authorities wished to reinforce the preventive actions regarding nuclear power plants. In this way, the ASN and the General Directorate for Health asked EDF, on the 28th of January 2005, to periodically supervise the concentration of Legionnaire's bacteria in the cooling towers (supervision that was already made by the operator). They also prescribed preventive measures for maintaining the level below the fixed levels, and the arrangements to be taken by the operator in case these levels are exceeded.

Equipment and installations similar to conventional cooling industrial installations, for instance, and which are not necessary for the operation of the BNI, are covered by a regulation, which is specific for installation classified on environmental protection grounds.

**Art. 7 – Legislative and Regulatory Framework**

| Seq. No 26 | Country | Article Article 7 | Ref. in National Report |
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| Question/ Comment | Which formal procedures are foreseen to update the national regulatory framework continuously in accordance with the newest state of science and technology? How are recommendations issued by international institutions incorporated into the national regulatory framework? |
| Answer | The National regulatory framework is regularly updated by the French Nuclear Safety Authority (ASN). ASN drives its modifications from the outputs of its participation to national and international workshops and of its national and international watch.<br>Besides the regulatory framework, licenses are updated every ten years, on the basis of a global confrontation of the conception and fabrication hypothesis to the newest state of technology (periodic safety review). This periodic safety review insures the highest level of safety.<br>ASN widely collaborates to the European harmonization works led by the WENRA association. This work is devoted to determine common standard levels of safety, which would be incorporated by 2010 within the French regulatory framework. |

| Seq. No 27 | Country | Article Article 7 | Ref. in National Report 7.2&7.3, p25 to 36 |
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| Question/ Comment | The current status and scope of Legislation, Decrees, Ministerial Orders and Regulations related to Nuclear Installation construction, operation, decommissioning and Emergency Planning appear to be in a considerable state of flux (reference sections 7.2, 7.3, 15.1, 16.5.4) - apparently related to the aftermath of the 2002 consolidation of responsibilities for Nuclear Safety and Radiation Protection with ASN. The ASN Basic Safety Rules (RFS - see Appendix 2 - section A.2.2) seem to be equivalent to Regulatory Standards or Guides. There does not appear to be a framework for the Basic Safety Rules, they appear to be stand alone ad- hoc documents, hence it is difficult to see if there are important issues not addressed.<br><br>Please elaborate as to whether France would consolidate the basis for Regulating Nuclear Power Plant Operation into a single piece of legislation to clarify and rationalize the apparent complexity of the current situation referencing such a wide variety of specific and non-specific legal instruments and amendments (see Appendix 2) and such a wide diversity of responsible consultative bodies and committees (see section 8). |
| Answer | ASN regularly updates its regulatory framework (see below).<br><br>A new legislative Act has been prepared a few years ago, which is under consideration by the French Parliament.<br>Concerning regulation existing only as "Ministerial decisions", ASN will update its regulatory framework by 2010, especially in order to harmonize its regulatory framework according to the ongoing WENRA harmonization work.<br><br>ASN is also currently reorganizing its Basic Safety Rules (RFS) as Guides, within a structure more directly inspired by IAEA standards. |

Read also answer to question No 26:
The National regulatory framework is regularly updated by the French Nuclear Safety Authority (ASN). ASN drives its modifications from the outputs of its participation to national and international workshops and of its national and international watch.
Besides the regulatory framework, licenses are updated every ten years, on the basis of a global confrontation of the conception and fabrication hypothesis to the newest state of technology (periodic safety review). This periodic safety review insures the highest level of safety.
ASN widely collaborates to the European harmonization works led by the WENRA association. This work is devoted to determine common standard levels of safety, which would be incorporated by 2010 within the French regulatory framework.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 28 | | Article 7 | P23.Ch7 |

Question/ Comment: Does the safety authority utilize probabilistic risk assessment when reviewing and approving waivers? What is the process of approving waivers and is the waiver valid for specific plant or specific reactor type?

Answer: As mentioned in the report (§19.2.2, p. 118; §19.3.1, p. 124 and §19.4.1.2.2, p. 128) any waiver from the Technical Operating Specifications must be exceptional.

Depending on the situation, a waiver can be plant or reactor type specific and generally speaking, the process of it's approval is as follows:

1. the licensee asks for a waiver and provides the following elements to support his demand :
   - the reasons and motivations for departing from the rule,
   - an analysis of the deviation consequences on safety,
   - countermeasures to limit the consequences,
   - the scheduled outage time.
2. ASN approves or rejects the waiver based on the examination of the licensee's risk analysis.

In this regard, PSA is an element among others that is used by ASN when reviewing a waiver in order to make sure that the increase in core damage frequency given the unavailability considered is limited, taking into consideration any palliative measures that the operator plans to implement.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 29 | | Article 7 | Art. 7.3, P.28 |

Question/ Comment: On what basis does the Nuclear Safety Authority plan its licensing activity for the next year?

Answer: ASN activities are planed by a three-years strategic plan. This mid-term plan is updated every year.

Each division within ASN, central and regional divisions, annually prepares a plan inspired by the mid-term annual plan. These division's plans are discussed by December and validated by ASN Steering Committee.

ASN's licensing activity planning is a part of these division's plans.

| Seq. No 30 | Country | Article Article 7 | Ref. in National Report Paragraph 7.3.1.4 |
|---|---|---|---|

Question/ Comment Is ASN satisfied that safety cases for all PWRs in France have adequately addressed the issue of corrosion near penetrations of the upper head of the reactor vessel, as found in the Davis-Besse NPP in 2002?

Answer Firstly, it is worthwhile to note (as mentioned in the report §14.1.3.3, p. 69) that as soon as 1989, that is two years before the only CRDM nozzle leak having occurred in France (and 13 years before the Davis Besse event), the ASN had already discovered the issue of the possible leak in penetration of reactor vessel heads and had requested an "Inconel Alloys Zones review" on all reactor from the national operator EDF.

This review concerned all the nickel-based alloy components and parts of the main primary circuits, including vessel head penetrations. After the discovery of a leak during Bugey unit-3 hydraulic test, not only EdF replaced all the vessel heads with Inconel 600 according to the defects found in inspection, but also the ASN requested EDF to perform a proactive assessment.

Then, the ASN defined new regulation rules in its ministerial order of November 1999. It also published in March 2001 a decision asking the utility to define a global strategy for periodic inspection of these zones and other priority Inconel zones and the replacement in due time of vessel heads. This strategy is approved by the ASN. A report is provided each year by the operator to the authority resuming the main controls realized during the previous year and their results. The strategy is then adapted according to the proposal of the utility or the request of the ASN. The regulatory hydraulic test showed its importance during the 10-yearly outage.

| Seq. No 31 | Country | Article Article 7 | Ref. in National Report Paragraph 7.3.2.4.1 |
|---|---|---|---|

Question/ Comment Can ASN indicate the number of significant incidents which came to its attention in 2003? Also what in ASN’s view was the most serious incident which came to its attention?

Answer Q1/

To specify what is mentioned in the report (§19.4.1.4, p. 130) it can be added that for the only year 2003, 653 incidents were rated on the INES scale, among which 503 for safety, 166 for radioprotection, and 2 regarding releases to the environment. One event was rated at level 2 of INES scale and 104 incidents were rated at the level 1 - that is a proportion around 15%, the other being rated at level 0. Events reported for the protection of environment but not related to safety or radiological risk, were not rated on the INES scale; this concerned 56 events.

Q2/

The most serious event in 2003 was rated at level 2 and was about the risk of filter clogging in the water recirculation sumps, which was generic of all NPP.

Indeed, studies undertaken at the international level and research work recently carried out in France by IRSN raise interrogations on the possibility of filter clogging in the water recirculation sumps, located at the bottom of the reactor building. This clogging could compromise reactor cooling during some accidents.

These sumps collect water in the event of important leakage from the primary circuit during an accidental situation, in order to return it in the safety injection circuit and ensure reactor cooling.
Taking into account the potential impact that this phenomenon could have for safety, the ASN decided to rate it at level 2 on the INES scale.

| Seq. No 32 | Country | Article<br>Article 7 | Ref. in National Report<br>p.10 & p28-29 |
|---|---|---|---|

Question/ Comment

Reference: 2.3.6 and 2.3.7
Q1: The ASN together with EDF continues to improve the nuclear installation safety from lessons learned and from introducing new technologies. If I may call it ‘back-fitting’, do you have some legislative request(s) to do the ‘back-fitting’?

Answer

As indicated in the report (page 68), in the French regulation system, there is the 11 December 1963 decree on BNIs that, in its article 5, addresses this issue in relation to Periodic safety review and consequent modifications. There are also provisions in the 3 September 1979 Ministerial letter on safety measures to apply on 900 MWe and 1300 MWe reactors that require that conclusions drawn from construction and operation experience, along with lessons learnt from incidents or accidents occurred in France or abroad, from research programme conducted in France or abroad, be taken into account in order to enhance the safety level of NPPs.

Moreover, in the framework of the harmonisation work within WENRA (Western European Nuclear Regulators’ Association), it is planned to add new requirements in our regulation system to cover more specifically the issue of back-fitting.

| Seq. No 33 | Country | Article<br>Article 7 | Ref. in National Report |
|---|---|---|---|

Question/ Comment

Reference: 7 Legislative and regulatory framework;
“the Directorate General for Nuclear Safety and Radiation Protection is responsible, within its field of competence through (12.) contributing to information of the public on issues related to nuclear safety and radiation protection”
7.3.3.2 Formalisation of ASN decisions and formal notices;
“both decisions and formal notices are made public, notably via the ASN web-site (www.asn.gouv.fr)”

Q1: By watching the above mentioned ASN web-site, one can easily grasp that events/failures with the INES rating are reported about 700/year and information notes are issued about 15/year in average. Regarding the events information, we would like to know more information e.g. the root-cause of the events. Do you have a future plan for the public of disclosing more detailed events information?
Q2. It is also reported in 7.3.3.2 of formalisation of ASN decisions and formal notices that when a particular site is concerned, the Local Information Committee (CLI) is informed. Would you explain about the member of the CLI (number and what kind of people they are) and about the information disclosure system in the CLI?

Answer

Q1:

We acknowledge that, at least for a non-French speaking reader, to the contrary to the statement included in the question, it is absolutely not "easy to grasp" on the ASN website that information is indeed given to the public for the about 700 events rated each year on the INES scale.

Indeed English versions of event information notes (the only ones "easy to grasp", since they are mentioned on the front page) are issued only around 15/year since they are limited to INES level 2 events and to events having impact on foreign countries, i.e. transport events and lost sources (they are also reported to IAEA NEWS system).

In addition to this "international information", information about each of the events rated at INES level 1 is also given (in French!) on our website under the concerned facility: for instance, for the year 2003, and limiting to only PWR nuclear power reactors, 104 event reports can be found (13 generic events information and the 91 other events information located under the 19 EDF sites). Finally, as regards the events rated at INES level 0, there is no specific information notice (since there are of low importance and medias do not show any interest on them) but all the related information can be found inside the about 700/year follow-up letters from basic nuclear installation inspections (where it is discussed together with other topics) which are also made available to the public since 2002: these follow-up letters are found (in French!) on our website under the dedicated part to each of the Basic nuclear installation.

All together that makes each year around 3,000 pages of information available to the public, which are related to events and anomalies occurring in French Basic Nuclear Installations. In addition ASN answer to any specific supplementary information requested by the Public (mainly through the Local Committees for Information – CLI).

Q2:
Local Information Committees (CLI) are made of elected people from the General Council (Assembly governing the Local Department), mayors and concerned representatives, members of environmental protection associations and representatives from socio-economic actors. Representatives from the nuclear operators as well as from the ASN attend the meetings.

The detailed composition of the Committee and the number of members vary from one site to an other (typically from 20 to 70 people). Generally elected people (mayors, local and national representatives) represent half of the Committee and one of them chairs the Committee.

The working methods are various: some Committees have set-up working groups specialised on specific topics, other meet only in plenary sessions.

Information provided to CLI comes from one side from the operators, which provide regularly reports on the operation of their installations, on events or incidents having occurred, on important works implemented and on the other side from Public Authorities and notably from the ASN. CLI members may go with ASN inspectors during some inspections. In addition local ASN representatives (Regional Divisions for Nuclear Safety and Radiation Protection) keep the CLI chairman of any decision related to the nuclear installation likely to be of interest for it.

| Seq. No<br>34 | Country | Article<br>Article 7 | Ref. in National Report |
|---|---|---|---|

Question/ Comment What is the exact criteria which you use to distinguish power reactor and research reactor ?
What's the difference in licensing procedure, technical safety standards and regulatory inspection between the two?

Answer In France there is no legal distinction between power reactors and research reactors since they are both legally defined as Basic Nuclear Installations (BNI) and are subject to the two same types of regulations:
- licensing procedures;
- technical rules.

The licensing procedures are defined by decree 63-1228 of 11 December 1963 and apply equally to nuclear power plants and to research reactors. This decree notably provides for an authorisation decree procedure followed by a series of licences issued at key points in the plant’s lifetime: fuel loading or pre-commissioning stages, commissioning, final shutdown, dismantling. The technical nature of the BNI is mentioned in the authorisation decree.

The general technical regulations, based on article 10a of the previously mentioned decree of 11 December 1963, currently cover four major subjects: pressure vessels, quality organisation, BNI water intake and effluent release and external hazards and detrimental effects related to BNI operation. Some new orders are on preparation. The orders apply to all BNI.

The Nuclear Safety Authority (ASN) also issues Basic Safety Rules (RFS) on various technical subjects, concerning both PWRs and other BNIs. These rules constitute recommendations defining the safety aims to be achieved and describing accepted practice the ASN deems compatible with these aims. There are currently about forty Basic Safety Rules. They are not, strictly speaking, regulatory documents. Rules laid down in this context are particularly flexible, allowing for technical advances and new know-how, and may apply specifically to power reactors or to BNIs other than power reactors (e.g. RFS on the confinement).
The regulatory inspection is of the same nature for both types of BNIs.

| Seq. No<br>35 | Country | Article<br>Article 7 | Ref. in National Report<br>A7.3.2.1.2 P33 |
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Question/ Comment The data of inspections by ASN shows that number of unannounced inspections has increased up to 25% of total inspections by the year 2003. Further, the number of inspections in “Fire Protection” area during the year 2003 are forty six (46) which are more than double the inspections performed in any other inspection area during a year. France may elaborate:
• What is the rationale for gradually increasing unannounced inspections?
• Does this increase in number of unannounced inspections is a part of enhancing regulatory effectiveness or these are reactive inspections in response to specific incidents/situations?
• How the licensees view this inspection strategy of ASN regarding preparation for the inspections and availability of necessary resources for the inspection, mutual trust in the best interest of safety?

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| | • What has led to increased number of inspections in fire protection, a safety related system? |
| Answer | A minimal periodicity of inspection, according to topics of a definite list, is required for each BNI and each nuclear site, called “hard core”. An annual inspection programme is determined by the ASN. It takes into account, inspections already carried out, DRIRE and ASN information on various plants and progress made on technical subjects under discussion between the ASN and the operators. It is prepared using a methodical approach defining the hard-core, annual priority national topics and suitable coverage of the different sites.<br><br>This programme is not communicated to BNI operators. It includes BNI’s name, inspectors’ name, topic and announced or unannounced characteristic. Some topics may preferably be the subject of unannounced inspection, such as work site inspections during outages, solid or liquid waste, fire protection, radiation protection, and emergency preparedness. Other topics need licensee specialists to be present: such inspections are announced.<br><br>There is no legal basis defining the ratio between announced and unannounced inspection. The ratio generally lies between 15 and 25%.<br><br>On the topic of fire hazards, the Nuclear Safety Authority carries out a large number of inspections every year, based on prevention, installation design and fire-fighting. It should be noted that the order of 31 December 1999, which specifically applies to basic nuclear installations, in particular regulates the fire protection aspect and sets environmental protection requirements (retention of fire fighting water, containment of toxic fumes, etc.). That is why the operator decided in 2000 to set up a comprehensive Fire Protection Plan, the implementation of which has to be checked.<br><br>In this connection, the Nuclear Safety Authority continues to be vigilant regarding progress application of proposals for modernising BNIs in response to these new requirements and through an increased number of fire drills performed during unannounced inspections. |

| Seq. No | Country | Article | Ref. in National Report |
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| 36 | | Article 7 | p. 28-29 |

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| Question/ Comment | The National Report fails to mention the document, which regulates the conduct of safety review. Is there a document, which defines the requirements to the safety review report and how is the quality assured while performing the review? |
| Answer | There is not a specific document regulating the conduct of safety reviews. The provisions for such a review are given in several texts, which are mentioned in the report and whose the purpose are broader than the only safety reviews.<br><br>Firstly, the 10 August 1984 order on quality (see report §7.3.1.3, p. 29) provides a general framework for provisions to be taken by any BNI operator to produce, obtain and maintain plant and operating quality standards compatible with safety requirements. This order is applicable to the studies performed in the frame of the periodic safety review.<br><br>As mentioned in the report (see report p. 68) the principles that regulate the conduct of periodic safety review are defined in the 11 December 1963 decree on BNIs (article 5). Then, on a case-by-case basis ASN issues letters to the BNI |

operators that define the scope of the safety review and specific issues that have to be considered.

For example, for 30-year safety review for 900 MWe, as mentioned in the report (§14.4.1.3, p. 74), the ASN issued a letter in October 2003 instigating the safety review, determining the scope and the limits of the studies to be made by EDF, together with the deadlines to be met to enable the resulting modifications to be integrated on the 900 MWe reactors during their third ten-yearly outages scheduled as of 2008. This letter has been made public on ASN's internet site.

Read also here the answer to question n°96:
Hereafter are given the headlines of issues to be considered:

1/Internal and external hazards
- o simultaneous failure of equipment non designed to withstand seismic conditions
- o consideration of internal flooding in shutdown states
- o internal explosion
- o fire
- o seismic verification approach
- o Adverse weather conditions
- o Hydrocarbon slick drift on river or seaside

2/Accident studies and radiological consequences
- o cold overpressure
- o Long term phases assumptions for accident studies
- o Steam generator tube rupture
- o Severe accident radiological consequences
- o Containment
- o Beyond design basis equipment
- o Backup of Auxiliary Feedwater System tank
- o Post accident surveillance information

3/Design of systems
- o Design verification of civil engineering structures
- o Functioning of Plant Radiation Monitoring system
- o Reliability of heat removal system of the fuel building
- o Performance of safety injection system
- o Reliability of emergency cooling recirculation function

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 37 | | Article 7 | section B, p 27 |

Question/ Comment: As we understand the subsection 7.2.2.2 only basic safety rules are not legally binding, while all others are mandatory.
Could you provide the hierarchy of legislative framework (law, decree, ministerial order, safety rules) and who issues them?

Answer: As in a number of European countries the French nuclear regulation is basically a non-prescriptive one, the regulation being mainly focussed on objective to be met and leaving to the operator the choice of option to comply with. Therefore the strictly binding regulation is rather small but of high level (orders, decrees). The hierarchy of legislative framework is the following:

laws (issued by the Parliament),
decrees (issued the President of the Republic or the Prime minister),
orders (issued by one or several ministers or by the prefect for local issues),
basic safety rules, procedures or letters on safety rules (issued by ASN).

However, some ministerial orders may be signed by the Director General of ASN on behalf of the Ministers in charge of Nuclear safety and radiation protection.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 38 | | Article 7 | 7.3.2.1.2,part B |

Question/ Comment The section on Inspection Activities in Article 7 shows a table with the inspections performed by ASN from 1998 to 2003. Although it may be appreciated that the total number is practically the same in those two years, the so-called unannounced inspections increase from 68 to 176. To what is this due?

Answer Unannounced inspections have appeared to be the best way to check on the field, and on a day-to-day basis, how the operators deal with safety in normal operation without behaviour change due to the announced presence of inspectors.

See also answer as to first part of Question No 35:

A minimal periodicity of inspection according to topics of a definite list is required for each BNI and each nuclear site, called “hard core”. An annual inspection programme is determined by the ASN. It takes into account, inspections already carried out, DRIRE and ASN information on various plants and progress made on technical subjects under discussion between the ASN and the operators. It is prepared using a methodical approach defining the hard-core, annual priority national topics and suitable coverage of the different sites.

This programme is not communicated to BNI operators. It includes BNI’s name, inspectors’ name, topic and announced or unannounced characteristic. Some topics may preferably be the subject of unannounced inspection, such as work site inspections during outages, solid or liquid waste, fire protection, radiation protection, and emergency preparedness. Other topics need licensee specialists to be present: such inspections are announced.

There is no legal basis defining the ratio between announced and unannounced inspection. The ratio generally lies between 15 and 25%.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 39 | | Article 7 | page 32 |

Question/ Comment You write that annual inspections are either announced to the operator a few weeks beforehand or may be unannounced. What is the legal basis?

Answer Same Answer as to first part of Question No 35:

A minimal periodicity of inspection according to topics of a definite list is required for each BNI and each nuclear site, called “hard core”. An annual inspection programme is determined by the ASN. It takes into account, inspections already carried out, DRIRE and ASN information on various plants and progress made on technical subjects under discussion between the ASN and the operators. It is prepared using a methodical approach defining the hard core, annual priority national topics and suitable coverage of the different sites.

This programme is not communicated to BNI operators. It includes BNI's name, inspectors' name, topic and announced or unannounced characteristic. Some topics may preferably be the subject of unannounced inspection, such as work site inspections during outages, solid or liquid waste, fire protection, radiation protection, and emergency preparedness. Other topics need licensee specialists to be present: such inspections are announced.
There is no legal basis defining the ratio between announced and unannounced inspection. The ratio generally lies between 15 and 25%.

| Seq. No 40 | Country | Article Article 7 | Ref. in National Report 7.2.2.3 p.28 |
|---|---|---|---|

Question/ Comment: The National Report states that ASN will reach its conclusions on the 2000 edition of the RCC-M code (concerning mechanical equipment) in 2004. What conclusions have been reached on this revised industry code for mechanical equipment?

Answer: Contrary to what was expected at the time of writing the 3rd CNS report, the process engaged by AFCEN of bringing the French RCC-M code into line with the European ETC-M code is not still completed. Therefore ASN has delayed its conclusions, which will be published afterwards.

| Seq. No 41 | Country | Article Article 7 | Ref. in National Report 7.3.2.1.1 p.32 |
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Question/ Comment: The statement that ASN inspections are "neither systematic nor exhaustive" is somewhat surprising. There may be something lost in the translation here, which is perhaps intended to convey the idea that inspection by the regulatory body is always a sampling process which can never be guaranteed to detect 100% of non-compliances by the licensee. Would France please expand and clarify what was meant by the statement that inspections are "neither systematic nor exhaustive"?

Answer: Firstly it may be recalled that the principle of responsibility (set in Article 9 of this Convention) states that responsibility for hazardous activities lies primarily with those performing them (operator) and not with the public authorities or other parties.
In that sense, as suggested in the question, an ASN inspection consists in checking with a "sampling process" the observance of the regulation and the applicable technical documents, and in addition to make sure that the operator exerts his full responsibility. So it can be said that an inspection is neither systematic nor exhaustive: the inspector team never expect to check a 100% of compliances by the licensee.

| Seq. No 42 | Country | Article Article 7 | Ref. in National Report 7.3.2.1.2 p.33 |
|---|---|---|---|

Question/ Comment: The table on this page shows a significant increase in the proportion of unannounced inspection performed by the ASN. What are the reasons for this significant increase? Have unannounced inspections been found to be more effective? Are they more efficient? On what topic are the unannounced inspections concentrated?

Answer: Unannounced inspections have appeared to be the best way to check on the field, and on a day-to-day basis, how the operators deal with safety in normal operation without behaviour change due to the announced presence of inspectors.

See also answer as to first part of Question No 35:

A minimal periodicity of inspection according to topics of a definite list is required for each BNI and each nuclear site, called “hard core”. An annual inspection programme is determined by the ASN. It takes into account, inspections already carried out, DRIRE and ASN information on various plants and progress made on technical subjects under discussion between the ASN and the operators. It is prepared using a methodical approach defining the hard core, annual priority national topics and suitable coverage of the different sites.

This programme is not communicated to BNI operators. It includes BNI’s name, inspectors’ name, topic and announced or unannounced characteristic. Some topics may preferably be the subject of unannounced inspection, such as work site inspections during outages, solid or liquid waste, fire protection, radiation protection, and emergency preparedness. Other topics need licensee specialists to be present: such inspections are announced.
There is no legal basis defining the ratio between announced and unannounced inspection. The ratio generally lies between 15 and 25%.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 43 | | Article 7.1 | page 25 |

Question/ Comment: You write that France will get a new law on transparency and security in the nuclear field. When will it enter into force? How will it handle delicate questions like: how to deal with business secrets; how to decide which information must be protected and which information must be open for the public?

Answer: The draft law on transparency and security in the nuclear field is to be discussed within the French Senate in 2005.
It will handle conflicts with commercial and security secrets by restricting transparency to information unrelated to them. Such a restriction currently exists within French legislative framework, as, by Law, any French citizen is empowered to ask for the transmission of any document produced by the Administration.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 44 | | Article 7.2.1 | Sect. 7.2.2.1 pg. 27 |

Question/ Comment: In Section 7.2.2.1, General technical regulation, it is mentioned that the current body of general technical regulations will soon be changing, as the DGSNR is working on “broadening the scope of application.” It follows that three orders are under preparation, one of which involves nuclear pressure vessels. Please provide details about this order concerning nuclear pressure vessels.

Answer: The order in preparation concerning nuclear pressure vessels, mentioned in Section 7.2.2.1 of the report, is about equipment, which is especially built for nuclear applications. The objective of this order is to describe the conditions of the design, the manufacturing, the implementation and the operation of this equipment. The design and manufacturing of main primary system and secondary system are a specific issue of this text, while the operation of these systems is regulated by the ministerial order of November 10th, 1999.

| Seq. No 45 | Country | Article Article 7.2.3 | Ref. In National Report p.32 |
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Question/ Comment

Reference: 7.3.2.1 Inspection 7.3.2.1.1 Principles and objectives
In inspecting, please let us know about relation with risk information.
Q1: When you conduct inspection, how do you apply risk information to your inspections?
Q2: Do you use risk information in the inspection planning and the assessment of inspection results?

Answer

Q1/ ASN does not apply "risk information" – with the common understanding of PSA results - in its inspections.

Q2/
ASN does not use "risk information" – with the common understanding of PSA results - in its inspection planning neither in the assessment of inspection results.

An ASN inspection consists in checking that the operator complies satisfactorily with safety and radiation protection provision requirements. It is neither systematic nor exhaustive and its purpose is to detect specific deviations or non-conformances together with any symptoms suggesting a gradual decline in plant safety. They mostly take place on nuclear sites, but may also be carried out at operator Corporate engineering offices, at the workshops and design departments of a subcontractor or at the construction sites or at factories and workshops where various safety-related components are manufactured.

A large number of nuclear plant systems contain pressurised fluids and are consequently subjected to the general pressure vessel regulations. As regards the central government authorities, application of these regulations is monitored by the ASN for pressure vessels containing radioactive product. Among the pressure vessels within the scope of ASN supervision, the main primary and secondary circuits of the 58 EDF PWRs are of particular importance and their in-service behaviour is one of the keys to nuclear power plant safety. ASN supervision of these systems is consequently very specific. It is based:

– for the design and construction stage, on the ministerial order of 26 February 1974, concerning the main primary system, and on Basic Safety Rule II.3.8 (1990), concerning the main secondary system;
– for the operating stage, on the ministerial order of 10 November 1999, covering requirements for both these systems.

Pressure vessel operation is covered by supervision particularly focused on non-destructive tests, maintenance operations, the handling of non-conformances affecting these systems and periodic requalification of them.

| Seq. No 46 | Country | Article Article 7.2.3 | Ref. in National Report p.134 |
|---|---|---|---|

Question/ Comment

Reference : 20.2 International co-operation measures
Q1: Regarding the IAEA Safety Standards, do you have in the ASN some system e.g. a committee in order to investigate and/or implement them into the French regulations?

Answer

ASN executives do participate to the IAEA safety standards Committees and

Commission. IAEA Safety standards are therefore well-known and are generally in line with the French regulation. Their recommendations are implemented within the ASN organization or within the French national regulatory framework, as it is updated.

Even though no formal committee within ASN investigates today the implementation of safety standards, a reinforced structure for national regulatory framework updating is under study.

Meanwhile, ASN is preparing itself for an IRRT mission. The IRRT project self-assessments will obviously lead to the reinforcement of the implementation of organizational safety standards within French regulatory system.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 47 | | Article 7.2.3 | Sect. 7.3.1.1 pg. 29 |

Question/ Comment: It states that on-the-spot supervision (inspection) is provided at the nuclear plants by personnel from the Regional Divisions (DSNR) of the Directorate General for Nuclear Safety and Radiation Protection (DGSNR). It follows that DSNR personnel are in constant contact with the nuclear operators, and after step-by-step observation of the maintenance and refueling outages, the ASN has to grant authorization for restart. Please discuss this authorization process further. Is there a written documentation that accompanies this? Whose input is it based upon? And at what level in the ASN is this process carried out? (Pg. 35, Section 7.3.4.2)

Answer: Four months before a maintenance and refuelling outage, the plant operator sends to the ASN, and to its Technical Support Organisation IRSN, the planning of the maintenance operations programmed on the primary and secondary circuits.

ASN validates this program, after checking it includes the necessary maintenance. During the outage, DSNR inspectors follow the good implementation of the maintenance planning. They randomly inspect on-the-spot the basic maintenance operations, and they almost systematically inspect the major maintenance operations (removal of steam generators, for example).
All operations on primary and secondary components lead the operator to product:

- non-destructive controls reports ;
- calculation notes on the kinetics of development of the detected defects.

These documents are reviewed and validated during the outage by ASN (DSNR inspectors, DGSNR divisions, mostly BCCN), and IRSN.

After refuelling, the transition above 110°C (Celsius) of the primary circuit requires the ASN approval under the November 11, 1999 Ministerial order. The plant operator must ask for this approval before this transition.

A 1982 ASN decision states that ASN authorizes the reactor's nuclear divergence, after examination by IRSN, DSNR and DGSNR divisions of the on-the-spot inspections results, control reports and completeness of the maintenance planning. The authorization is granted by the Director General or his deputies, after proposal by DSNR, which includes IRSN expertise, and validation by DGSNR's reactors division.

**Art. 8 – Regulatory Body**

| Seq. No 48 | Country | Article<br>Article 8 | Ref. in National Report |
|---|---|---|---|
| Question/ Comment | The number of 312 staff (equalling 5 staff per NPP unit) in the French Nuclear Safety Authority is a rather low number. What are the reasons for this? | | |
| Answer | The French Nuclear Safety Authority (ASN) is the nuclear regulatory body. ASN inspects, authorizes and controls radiation protection and nuclear safety in France. In addition to this, as mentioned in the report (§8.1.4.1, p. 42), a Technical Support is provided to ASN by the Institute for Radiation protection and Nuclear Safety (IRSN). The IRSN, where most French nuclear safety experts are gathered, counts around 1500 staffs.<br><br>On most technical affairs, ASN will ask for the expertise of IRSN. Therefore, French staff devoted to nuclear safety and radiation protection is slightly lower than 2000 people.<br><br>Further more, ratio on the single NPP units may not be the most significant indicator as on one hand ASN also controls nuclear research laboratories, fuel cycle facilities and most of the medical and industrial activities where radioactive materials are implemented, but on the other hand NPPs are grouped, mostly by four units, on a small number of sites therefore allowing supervision resources to be used in common. | | |

| Seq. No 49 | Country | Article<br>Article 8 | Ref. in National Report<br>8.1 |
|---|---|---|---|
| Question/ Comment | 1. What does the term "the latter" in the second paragraph of Subarticle 8.1 stand for?<br><br>2. What is the background for merging the areas of nuclear safety and radiation protection in a single regulatory body?<br>Did you evaluate the outcome of reorganization of regulatory authority in 2002? If yes, what are the results and the lessons learned? | | |
| Answer | Q1:<br>"the latter" means "the DGSNR" in this paragraph.<br><br>Q2:<br>The merging of nuclear safety and radiation protection in a single regulatory body contained in several decrees dated 22 February 2002 is consistent with the previous proposals by deputy J.-Y. Le Déaut, then Chairman of the Parliamentary Office for the Assessment of Scientific and Technological Options.<br><br>As it was discussed in details during the 2nd CNS Review Meeting, this reform aims to unify the supervision of nuclear safety and radiation protection, to bolster the resources devoted to supervision of radiation protection and to clarify the status of the former Institute for Nuclear Safety and Protection (IRSN), the technical support organization of the Nuclear Safety Authority.<br><br>Supervision of nuclear safety and of radiation protection was unified at two levels: in the regulatory organizations and in their technical support organizations. In | | |

terms of the regulatory authorities, the Nuclear Safety Authority which was in charge of monitoring nuclear safety in the basic nuclear installations, was merged with the various departments which regulated and supervised radiation protection: the former Radiation Bureau of the General Directorate for Health, a part of the former Office for Protection against Ionizing Radiation (OPRI), and most of the former Secretariat of the Interministerial Commission for Artificial Radioelements (CIREA), a commission which was in fact closed down at the end of March 2002. The new Nuclear Safety Authority was thus placed under the authority of three Ministers: the Minister for Industry and the Minister for the Environment, with respect to nuclear safety activities, the Minister for Health, with respect to radiation protection activities. In terms of the technical support organizations, the former Institute for Nuclear Safety and Protection (IPSN) was merged with the largest part of the former OPRI, to form the Institute for Radiation Protection and Nuclear Safety (IRSN).

No specific outcome was expected after the 2002 reorganization, besides the creation of a new inspectorate for radiation protection and the corresponding increase of the staff over the following years. Long-term outcomes are expected as ASN is now richer of more staff and two complementary missions (Nuclear Safety and Radiation Protection).

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 50 | | Article 8 | 8.1.2 p38 |

Question/ Comment: The National Report refers to the implementation during 2004 of possible changes in the supervision of radiation protection outside BNIs, with particular reference to two regions. What changes have been implemented, or are planned to be implemented?

Answer: Since it was created in 2002, the DGSNR has worked at organizing and developing the inspection of radiation protection outside BNIs. Identification of control priorities, definition of action procedures and deployment of the necessary workforce are all being carried out in parallel. The ASN is devoting attention to setting up an effective and well-proportioned supervision system, drawing on the experience of the personnel from the former permanent secretariat of the Secretariat of the Interministerial Commission for Artificial Radioelements (CIREA) and former Office for Protection against Ionizing Radiation (OPRI) who have joined it in 2002, and relying on the State's regional administration, whose actions in the field are under its responsibility. The ASN also listens closely to the parties concerned by the use of ionizing radiation and keeps an open mind with regard to foreign practices. The draft Act on nuclear transparency and security comprises provisions which will be such as to backup the regulatory tools in this inspection system, which will achieve maturity with the gradual addition of the one hundred and fifty inspectors.

With this aim in mind, the Director General for Nuclear Safety and Radiation protection decided that two DRIREs, in the Basse-Normandie and Rhône-Alpes regions, would carry out a « reconnaissance » mission, in order to initiate radiation protection control practices in non-BNI areas.

The major changes that have been implemented are:

- the creation of two new ASN regional divisions, one in Nantes for the Brittany and Pays de Loire areas and another one in Paris for the Ile-de-France area;
- the creation of a team of 90 radiation protection inspectors, distributed within ASN's regional divisions;
- the reinforcement of ASN's work with other administrations, especially those in charge of Health and Labour;
- the finalization of the implementation of the 97/43 and 96/29 European directives into the French regulation.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 51 | | Article 8.1 | p.37 |

Question/ Comment: Reference : 8.1 The Nuclear Safety Authority (ASN)

Q1: Is there the right of dismissal of the head of the ASN during the term of his/her appointment?

Answer: The head of the ASN, like any Director within the French central Administration, is named by decree of the French Republic President after report by Prime Minister and by the Ministers in charge of Health, of the Environment and of Industry. His mandate is granted for an indeterminate period.

The decision to name a new head to the ASN may be taken at any time through the same procedure.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 52 | | Article 8.1 | |

Question/ Comment: ASN uses for its technical expertise outside technical organizations. Could ASN mention if the ASN is the only beneficiary of these technical organization, if not, how the conflict of interest is avoided?

Answer: IRSN is the major Technical Support Organisation working for ASN. This Institute is an "industrial and commercial public establishment" (so-called EPIC). Its cash flow mainly comes from the State (ASN, other French safety agencies and the Research Secretary of State). Even through, it has the possibility to sell its expertise.

Conflicts of interest are avoided by a code of ethics, provided for by Article 18 of the 2002-254 decree which creates IRSN, which obliges to insure separation within IRSN between expertise provided for State Administration and expertise both performed for private or public companies.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 53 | | Article 8.1 | |

Question/ Comment: It is mentioned in the report that in France there are no resident inspectors. What are the means and arrangements made between ASN and NPPs in order to assure a daily surveillance of NPPs status and operation conditions?

Answer: The principle of responsibility (set also in Article 9 of this Convention) states that responsibility for hazardous activities lies primarily with those performing them (operator) and not with the public authorities or other parties.

In France, any inspector belonging to ASN's divisions (DGSNR or DSNRs) may inspect any BNI on the national territory. There is no resident inspector and all inspectors are based on ASN premises but each BNI has an inspector who his more specifically in charge of its supervision. This allows to have the support of specialist inspector for specific topics and to have some experience of practices of different nuclear site and licensees. Nevertheless, as either inspector or instructor, an ASN member is about 2 days per week on NPP site and ASN have daily contact by telephone or E-mail with the operator.

In addition it could be mentioned that unannounced inspections (which now represent 25% of the total) have appeared to be the best way to check on the field, and on a day-to-day basis, how the operators deal with safety in normal operation without behaviour change due to the announced presence of inspectors.

| Seq. No 54 | Country | Article<br>Article 8.1 | Ref. in National Report<br>P. 38/39Ch 8.1.2 |
|---|---|---|---|

Question/ Comment: From the report it is clear that the interface and the responsibilities between the DRIRE and the DGSNR remains the same as in the previous DSIN (except that the scope of the DSNR has been expanded in terms of the new mandate of the DGSNR). However what is not so clear is the interface (reporting lines) and decisions making power of the DRASS and DDASS in relation to the DGSNR. Can you please provide some additional information on this point?

Answer: In 2003 and 2004, a working group comprising representatives of the DRIRE, DRASS and DDASS was tasked with drawing up procedures for collaboration between the entities in this field.
The conclusions of this group were gathered within a guide, which precisely describes the responsibilities of DSNR and DDASS and DRASS on the field. Meanwhile, DGSNR became a recognized ministry directorate for DRASS and DDASS, within the French Health central Administration. For example, DGSNR coordinates the distribution by DRASS of stable iodine around nuclear plants.

| Seq. No 55 | Country | Article<br>Article 8.1 | Ref. in National Report<br>P.41 Ch. 8.1.3.2 |
|---|---|---|---|

Question/ Comment: Staff training:
In terms of the additional mandate given to the DSGNR ( as opposed to the DSIN) has the training programme of the inspectors being updated accordingly to cover all areas of the mandate or are there dedicated specific training programmes for BNI inspectors and for radiation protection inspectors? Can you please provide some additional information?

Answer: Radiation protection inspection required new legal basis, which was incomplete at the time of the creation of DGSNR and was finally enacted by appropriate law in August 2004. In parallel DGSNR is currently preparing a decree for the implementation of this law. It is awaited for the end of 2005. Pending official radiation protection inspections, ASN is currently organising "radiation protection reconnaissance visits", which provide only for recommendations without formal legal value but which are a strong incentive to correct the issues.

ASN has therefore already prepared a specific training program dedicated for future Radiation protection inspectors. Global training on specific topics such as quality is expected to be ready for BNI and Radiation protection inspectors. Today, to perform reconnaissance visits to the users of radiation, a two months training covers all areas of the radiation protection supervision mandate.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 56 | | Article 8.2 | Para 8.1, P.37 |

Question/ Comment: Considering that three ministries are above the Nuclear Safety Authority, how is its independence assured?

Answer: De jure, the multiplicity of ministers involved makes possible a multiplicity of opinions: this is a guarantee of independence. In case of irreconcilable discrepancy between opinions of the ministers, a meeting is held by the Prime Minister's services, which brings together ASN, and the services of the involved ministers and takes the final Government's decision.
De facto, ASN independence is observed in its day-to-day operation. Moreover its independence is acknowledged by most of French stakeholders and is not at all an issue.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 57 | | Article 8.2 | Sect. 8.1 pg. 37 |

Question/ Comment: The organizational structure of the Nuclear Safety Authority (ASN; regulatory body) is described. It is explained (with an accompanying diagram) that the ASN reports to both the Minister for Industry and the Minister for the Environment, and that “this double supervision guarantees the independence of the ASN from the Directorate General for Energy and Raw Materials,” which is responsible for nuclear energy development and which reports exclusively to the Minister for Industry. It is not clear how independence is maintained when both ASN and the Directorate General for Energy and Raw Materials both report to the Minister for Industry. Please explain this organizational relationship further.

Answer: To the difference of the Directorate general for energy and raw materials, which reports only to the Minister for Industry, ASN reports jointly to both the Minister for Industry and the Minister for the Environment as regards nuclear safety. It reports also to Minister for Health as regards radiation protection.

De jure, the multiplicity of ministers involved makes possible a multiplicity of opinions: this is a guarantee of independence. In case of irreconcilable discrepancy between opinions of the ministers about safety issues, a meeting is held by the Prime Minister's services, which brings together ASN, and the services of the involved ministers and takes the final Government's decision.

De facto, ASN independence is observed in its day-to-day operation. Moreover its independence is acknowledged by most of French stakeholders and is not at all an issue.

Therefore, ASN safety decisions are independent from energy policy decision taken by the Directorate general for energy and raw materials, which reports only to the Minister for Industry.

**Art. 9 – Responsibility of the licence holder**

| Seq. No 58 | Country | Article Article 9 | Ref. in National Report |
|---|---|---|---|
| Question/ Comment | The report states that the operator of a Basic Nuclear Installation (BNI) must ensure that quality is defined, obtained and maintained for the various components of the facility and their operating conditions, in accordance with the safety importance of their functions. Which procedures and checks are applied to guarantee that the licence holder fulfils its implied responsibilities? | | |
| Answer | This topic is mainly related to the August 1984 Quality Order discussed under Article 13 (and also 7) of this report. It is checked by the ASN during inspections at different levels of the licensee, including Corporate headquarters. The elaboration and implementation of the licensee's quality system is assessed during these inspections. It is supplemented by meetings and information exchanges between the ASN and the licensees. | | |

**Art. 10 – Priority to safety**

| Seq. No 59 | Country | Article<br>Article 10 | Ref. in National Report |
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| Question/ Comment | The report states, that EDF is of the opinion that safety and competitiveness (availability and costs) can only be improved if all employees of EDF are sensitized for their responsibilities. How are those principles handled in cases, where there is a conflict between the two goals? |
| Answer | On an everyday basis, the NPP safety engineers carry out independent safety analysis and compare each day's results with the analysis of the operating manager who is in charge of safety. Whenever a conflict appears, an arbitration request is made by site management. Experience feedback reveals that there are few cases where arbitration is necessary because these meetings make it possible to go deeper into safety analysis.<br><br>As stipulated in the report, the safety assignment chief, reporting to the site Management, also has the duty of directly warning the Corporate Management Safety Delegating Director if he estimates that the site safety decisions are not satisfactory.<br><br>What is more, site management implements OSRDE (Observatoire Sûreté Radioprotection Disponibilité Environnement, or SAREO, Safety Availability Radioprotection Environment Observatory) to check after the fact whether the decision-making process and the conditions under which safety / availability decisions were reached by arbitration were satisfactory. SAREO is an approach that was constructed by EDF but which is in line with the "Operational Decision Making" method used. |

| Seq. No 60 | Country | Article<br>Article 10 | Ref. in National Report<br>p. 47, 10.2 |
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| Question/ Comment | EDF installed a safety management system in 1997.<br>On which requirements has this been based (e.g. IAEA, ISO standards)?<br>Is a review of this system part of the ten-yearly visits (visite decentrale)? |
| Answer | EDF set up in 1997 an initial safety management outline built on the basis of the use of management tools that involved risk analysis, self-diagnosis, self-evaluation, a sensitive transient state approach, SAREO (Safety Availability Radioprotection Environment Observatory), and operational communication. Gradually the sites set up these tools in the department and progress was made on all of them.<br><br>During 2004, an extensive job of capitalizing on site methods was carried out; it made it possible to identify efficient methods on the site and showed the need to include the use of all these tools in a more global safety management framework. A "Safety Management Policy Application Guide" offered the opportunity of establishing the link between the safety policy, the Division quality management policy and the use of different tools. It is built on the basis of eight principles of EFQM which themselves structure the Division management policy published in 2004, October.<br><br>This is a reference document, presenting for each principle (controlled by results, leadership and constant vision, implication and development of the staff, brushes of |

apprenticeship and innovation…) methods confirming the approaches that can be implemented in the field of safety, as well as a limited number of prescriptions. Documents INSAG 4, 13 and 15 and on another level, the quality management approach (EFQM) are the founding documents of this reference document.
This action is not connected with ten yearly outages.

| Seq. No 61 | Country | Article Article 10 | Ref. in National Report p. 49, 10.2 |
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Question/ Comment: In chapter 10 safety indicators are mentioned. Please provide more details on the use of safety indicators for French plants. To what extent is the regulator involved?

Answer: Each NPP site has its own safety control system based on safety indicator scorecards, the result of internal and external assessment (EdF Nuclear Inspection, Nuclear Safety Authority, IAEA).

At national level, nuclear site safety performance evaluation is based on the measurement of the results (through a limited number of evaluated indicators, results of inspection, and on event feedback), and on the evaluation of managerial dynamism in terms of safety (based on the result of safety evaluations, product of safety reporting by site, hierarchical inspection on site by the Nuclear Production Division Managerial team.

Generally speaking, the evaluated indicators are only one part of the general safety performance evaluation elements. A greater number of indicators used for the control of the various processes. EDF will continue to improve and develop its safety performance evaluation system on nuclear sites in the years to come.

The Nuclear Safety Authority does not intervene in this management system. It carries out an evaluation that is independent of the one performed by the Operator, through its own system of inspection and the information supplied by the operator (and especially the event and deviation reporting and their processing).

| Seq. No 62 | Country | Article Article 10 | Ref. in National Report P51 |
|---|---|---|---|

Question/ Comment: Regarding the availability of sufficient number of qualified staff with appropriate education, training and retraining, the human resources are described in the Article. However, the regulator's role in assuring sufficient number of personnel having adequate competency for handling safety related activities has not been addressed. The data of inspections for the year 2003 presented in Article 7 of the report did not show any inspection activity related to training & qualification of the licensee's personnel during the year 2003. France may clarify that what are the regulatory activities regarding assessment of adequacy of training and qualification of the personnel which are assigned duties having bearing on safety?

Answer: Indeed, only a few inspection topics were mentioned in the part of the report devoted to Article 7. In 2003 and 2004, several inspections were conducted about training and qualifications of the licensee's personnel.

In addition, ASN has asked his Advisory Committee of experts for nuclear reactors to examine EDF's management of training and staff authorization at NPPs. The Advisory Committee's advice is expected for end 2005.

| Seq. No 63 | Country | Article Article 10 | Ref. in National Report |
|---|---|---|---|

Question/ Comment: What characteristics and indicators have been adopted in France for evaluating safety culture status?

Answer: Safety culture status is evaluated by the operator, essentially through two different elements:

- The event declaration system includes characterization through a number of criteria, one of which is relative to the safety culture.
- The sites are also provided with a tool known as the "Safety Perception Questionnaire "aimed at obtaining an evaluation of the safety perception of the site by its officials. This questionnaire was established in a particularly stringent manner and is foolproof from the methodological point of view. It is a way of taking a snapshot of the perceptions by profession, seniority, a hierarchical position etc. It can be advantageously used in an approach towards progress, to evaluate the effect of the implemented actions.

The safety culture is not directly regulated in itself, in the way the ASN do not measure a "level of safety culture" that could be obtained from people or team behaviours, attitudes, etc. It is more the process of managing the safety and improving it that is examined by the ASN, including the use by the licensee of appropriate levers, tools in this process.
The safety culture is considered by ASN in the analysis of events as described in the INES scale, throughout inspections dedicated to this topic and also throughout the transparency of the licensee with regard to his relation with the ASN.

| Seq. No 64 | Country | Article Article 10 | Ref. in National Report section 10.2, p 48 |
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Question/ Comment: It is mentioned that strategic control allows division management to ensure that the strategy guidelines system are appropriate and that sub-division project are in line with the policy reference system.
Does the division management perform analysis for sufficient and competent staff and resources through the life of a nuclear installation, considering pensioning off staff?

Answer: The recruiting of NPP site staff was particularly intense in the early 80s. The age pyramid of EDF Nuclear Production Division (NPD) staff is therefore particularly deformed and pension departures will begin acceleration in 2006, at different rates according to the professions.

The renewal of skills is a triple "challenge": succeeding in transferring to a new generation the skill acquired since start-up. Training times are particularly long for some professions and require real anticipation. This is an opportunity for the younger staff to develop professionally and for the company to optimize its organizations and skills.
In 2003, EDF therefore undertook a project known as the "Renewal of Skills" for which there are 5 goals:

- lighting the way towards the future by generating a prospective vision for each of the professions;

- setting up a method of prospective job and skills management within the NPD, in each production unit and each professional department, based on a map of employee proficiencies. This approach, destined to anticipate the necessary flows of personnel (re-deployment, hiring) is well advanced now.
- organizing the transfer of skills between those who are leaving and the newcomers: the necessary actions are now being put under way among all the units: integration of newcomers, training and gradual clearance, organization of apprenticeship and supervision, transfer of the more difficult and implicit knowledge fields.
- organizing redeployment towards the technical professions in the NPD identified as attractive professions within the company,
- focusing more particularly on the sensitive skills of nuclear engineering that have been clearly identified.

Skills in operation, safety control and nuclear maintenance are particularly important of which, this is why:

- for maintenance, the proficiencies are examined both internally and externally to establish the industrial strategies
- the deployment of a full-scale simulator on each production site was carried out in order to have a rugged system of training-professionalization.
- In the longer-term, establishing sufficient attractiveness among future technical professions on the employment market will be one of EDF focal points.

| Seq. No<br>65 | Country | Article<br>Article 10 | Ref. in National Report<br>page 47 |
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| Question/ Comment | How is the Safety Management System integrated in the overall Management System of the plant? What is the link to Quality Management? |
| Answer | EDF applies to safety the principles of its management through quality. The Nuclear Production Division wants to broaden the overall performance figures through safety performance achievements for which management must be exemplary.<br><br>Read also the answer to question No 60:<br><br>EDF set up in 1997 an initial safety management outline built on the basis of the use of management tools that involved risk analysis, self-diagnosis, self-evaluation, a sensitive transient state approach, SAREO (Safety Availability Radioprotection Environment Observatory), and operational communication. Gradually the sites set up these tools in the department and progress was made on all of them.<br><br>During 2004, an extensive job of capitalizing on site methods was carried out; it made it possible to identify efficient methods on the site and showed the need to include the use of all these tools in a more global safety management framework. A "Safety Management Policy Application Guide" offered the opportunity of establishing the link between the safety policy, the Division quality management policy and the use of different tools. It is built on the basis of eight principles of EFQM which themselves structure the Division management policy published in 2004, October.<br><br>This is a reference document, presenting for each principle (controlled by results, |

leadership and constant vision, implication and development of the staff, brushes of apprenticeship and innovation…) methods confirming the approaches that can be implemented in the field of safety, as well as a limited number of prescriptions. Documents INSAG 4, 13 and 15 and on another level, the quality management approach (EFQM) are the founding documents of this reference document.

This action is not connected with ten yearly outages.

**Art. 11 – Financial and human resources**

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 66 | | Article 11 | p. 52 |

Question/ Comment: On which assumptions is the budget for decommissioning based? Has a decommissioning plan and an appropriate fund been setup for this purpose? Is the annual budget for decommissioning fed into some kind of a decommissioning fund?

Answer: The budget for decommissioning is based on engineering studies for the decommissioning of each installation, wastes management including repositories, industrial capacities and existing regulations for decommissioning.

Complete decommissioning plan has been set up with corresponding agenda; there is no specific fund to finance such plan. EDF finance decommissioning plan under pluri-annual budget.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 67 | | Article 11 | P. 5-8 + p. 53 |

Question/ Comment: Today, EDF is facing strong competitive situation in electricity market and the operators are paying more attention to cost control. How do the nuclear power plants cope with the potential safety impact caused by cost control? What tools does ASN use for the early detection of potential safety problems mentioned above?

Answer: Generally speaking, the Nuclear Safety Authority and the Nuclear Operator are aware of the potential safety risk that a quest for competitiveness represents. The operator is more than aware of the fact that the future of nuclear energy development is only possible if faultless safety is provided for.

From the point of view of the operator, the experience of several years' research in competitiveness, which did not wait for the electricity market to open, reveals that there is no contradiction between safety performance and overall performance. To the contrary, it is observed that economic performance and safety performance generally go hand-in-hand.

EDF is therefore clearly posting safety to be its priority, which is translated into every level of corporate management in a concrete manner. At the global company level, several provisions have been implemented and among the most important are:

- The liabilities for nuclear safety are covered by management which, at each level, has a safety committee, made up of full-time safety specialists and an independent internal inspection system (Nuclear Safety General Inspection for Corporate management, at Nuclear Inspection for the Nuclear Power Division, Quality Safety Structure for the NPP sites). At the higher level, mandatory safety reports have been established.
- Budget arbitration does not concern safety files and the operator periodically reports to the Nuclear Safety Authority.
- The institution of periodic safety reviews in particular guarantees the maintained approach towards progress in terms of safety.

To cope with the potential safety impact caused by cost control, EDF developed a “cost /safety benefit” approach. The principle is to estimate in a part all the "costs" in Unity of work (days of unavailability, Work hour), then in euros, on the other hand the Safety impacts, if possible in a quantitative way, otherwise in a qualitative way (safety is relative to prevention and mitigation of accidents or all radiological concerns). A safety benefit ratio is then calculated for each potential plant evolution. Evolutions are then prioritized according to this ratio. This approach enables EDF to maximize safety increase within well-defined resources (human and financial). This approach has been developed transparently with the Nuclear Safety Authority to make engineering decisions more pertinent.

Through its current supervision of the nuclear operator’s internal safety management, the ASN analyses the way the nuclear operator takes its decisions and it looks after that safety remains its priority. This usual supervision, which is done at every level of the nuclear operator’s structure (corporate management, NPP sites…) is a tool used by the Nuclear Safety Authority to detect potential safety problems which could be caused by the fact that the nuclear operator is paying more attention to the cost control in a strong competitive situation.
The ASN has recently developed additional tools to early detect potential safety problems mentioned above. The ASN analyses every year the financial statements of the nuclear operator, expenses evolution, the number of employees and organisational changes. The ASN also analyses safety indicators trends and new optimisation programs, such as maintenance programs. In the future, the ASN will continue to think about new tools to detect potential safety problems caused by the response of the nuclear operator to the competitive environment.

Read also answer to question No 60 concerning safety management:

EDF set up in 1997 an initial safety management outline built on the basis of the use of management tools that involved risk analysis, self-diagnosis, self-evaluation, a sensitive transient state approach, SAREO (Safety Availability Radioprotection Environment Observatory), and operational communication. Gradually the sites set up these tools in the department and progress was made on all of them.

During 2004, an extensive job of capitalizing on site methods was carried out; it made it possible to identify efficient methods on the site and showed the need to include the use of all these tools in a more global safety management framework. A "Safety Management Policy Application Guide" offered the opportunity of establishing the link between the safety policy, the Division quality management policy and the use of different tools. It is built on the basis of eight principles of EFQM which themselves structure the Division management policy published in 2004, October.

This is a reference document, presenting for each principle (controlled by results, leadership and constant vision, implication and development of the staff, brushes of apprenticeship and innovation…) methods confirming the approaches that can be implemented in the field of safety, as well as a limited number of prescriptions. Documents INSAG 4, 13 and 15 and on another level, the quality management approach (EFQM) are the founding documents of this reference document.

This action is not connected with ten yearly outages.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 68 | | Article 11 | §2.3.1, P. 8 + p. 54 |

Question/ Comment: How can the Nuclear Safety Authority harmonize the cost-benefit approach with the „safety first” principle?

Answer: The "cost-benefit for safety" approach was presented by EDF to the ASN at the beginning of 2003 during the preparation of the third ten-yearly outages for the 900 MWe plants (See report, §11.4.1, p. 54). Currently, the ASN has not given any definitive position regarding this approach.

In any case, the resources of the licensee are limited. Thus, the "cost-benefit for safety" approach may contribute to a better allocation of resources on what has the most benefit for safety. Furthermore, "cost-benefit for safety" approach is not supposed to be used for non-conformances but only for safety enhancement.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 69 | | Article 11 | A11.4.1 P54 |

Question/ Comment: One of the areas the ASN has identified in the coming years includes the implementation of a more open and responsible dialogue with the operator concerning its economic issues. In this context, it is mentioned that the ASN is ready to examine a cost benefit argument in which the operator will demonstrate that some improvements asked for by ASN do not represent an optimum allocation of available resources and would propose devoting its resources to actions having a more beneficial impact on safety. In this approach, there is a dominant possibility that the licensee may always argue against the improvements required by the ASN having impact on plant availability. Since safety versus availability is always debatable, France may elaborate that how ASN will ensure conservative decision making with cost-benefit approach without compromising safety?

Answer: Same answer as to question No 68:

The "cost-benefit for safety" approach was presented by EDF to the ASN at the beginning of 2003 during the preparation of the third ten-yearly outages for the 900 MWe plants (See report, §11.4.1, p. 54). Currently, the ASN has not given any definitive position regarding this approach.

In any case, the resources of the licensee are limited. Thus, the "cost-benefit for safety" approach may contribute to a better allocation of resources on what has the most benefit for safety. Furthermore, "cost-benefit for safety" approach is not supposed to be used for non-conformances but only for safety enhancement.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 70 | | Article 11.1 | last para.,11.4.1, |

Question/ Comment: Reference : 11 Financial and human resources, page 54

Regarding harmonization of requirements: there exist already mutual utility requirements; EUR (European Utility Requirements) thus, from that experience, it may not be unrealistic to harmonize regulatory requirements.
Q1: Does the ASN have already a plan to develop harmonized regulatory requirements? How wide in the world the requirements will be applied to? How mane years are expected for developing the first draft?

| | |
|---|---|
| Answer | In 1999, ASN has contributed to the creation of WENRA (Western European Nuclear Regulators' Association) which now comprises 17 member states of the European Union and which is a forum for exchanging information and experience with a view to promote continuously improving safety and to harmonise its safety approach. ASN participates actively in the 2 working groups of WENRA, whose aims are harmonisation of nuclear safety for reactors, fuel cycle facilities, waste storage or dismantling.<br><br>The 17 member states of WENRA committed themselves to achieve an harmonised level of safety by 2010, that is to say to complete harmonisation of regulations and harmonisation of implementation on the facilities.<br><br>In 2004, the working group on waste and decommissioning issued a report on reference levels that are considered as harmonised regulatory requirements on that topic. The working group on reactors harmonisation is expected to issue a similar report by the end of 2005. |

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 71 | | Article 11.2 | p. 51, 11.2 |

| | |
|---|---|
| Question/ Comment | Are there any qualification and retraining requirements for the management personnel of the operating organisation?<br>Is the authority regularly informed about the respective activities? |
| Answer | The qualification and retraining requirements are included in the general regulation stated in the August 1984 Quality Order. More detailed requirements are developed by the operators, the implementation of which is assessed by the ASN through inspections.<br><br>There are several elements involved in selecting future NPP managers at EDF whose designation is handled by the management:<br>- Their initial training<br>- Their professional background ensuring their technical capability in the field concerned<br>- In some cases a national or local jury confirming technical proficiencies on the one hand, and managerial skills or potential on the other.<br><br>In the field of control in particular, the engineers intended to occupy an MPL – First Line Managers – function (operation manager, handing safety responsibilities in real time) have initial engineer's training and go through additional technical training before moving into an operational manager's position. In any case, they go through a specific training syllabus involving the acquisition of technical skills (safety) and managerial abilities.<br><br>Once the applicant has been validated to take a managerial position, there are professionalization actions:<br>- In every technical field concerned, depending on individual needs;<br>- From the managerial point of view: newcomers to a managerial function go through a professionalization course which is based on the Corporate reference skills and managerial activities. The courses are different according to the level of management: a First Line Manager course or a Second Line Manager course (head of Department), a Unit Director course. |

- Participation in the national and local networks in the areas concerned, allowing the updates needed and the exchange of methods between managers working in the same professional area.

Training days for newly promoted heads of department in the most prominent professions are organised at the Nuclear Production Division national level (Operation Department Managers, Safety Department Managers).

The CEA considers the installations safety level of prime importance and directly bounded to the qualification and training of managers.
Different training courses are so in place for all managers: heads, safety engineers, critician engineers.

For what concern the installations heads, a list of potential candidates is established based on the experience and motivation of the candidates. A special training for installations managers is in place for these candidates, covering all safety and organisation aspects. Complementary trainings were also developed on human factors and on incident analysis.

If necessary, these trainings are completed by a specific formation plan in particular fields.

More recently rules were established for the safety and critician engineers.
The Nuclear Safety Authority is informed of these actions through typical inspections.
All these general rules are completed by installation internal rules including a continuous formation plans.

**Art. 12 – Human Factors**

| Seq. No 72 | Country | Article Article 12 | Ref. in National Report 2.3.4, p9 + p. 58 |
|---|---|---|---|

Question/ Comment: In relation to the importance of human and organizational factors, the report indicates that "the ASN … has undertaken a methodical review …"
Please provide details of the main findings of such a review, and how such findings were dispositioned.

Answer: The ASN approach is methodical in the way the different topics relevant of the human factors fields are considered systematically. In 2004, the approach used for integrating human and organisational factors in the design and implementation of modifications of nuclear power plants have been evaluated. Findings concerned improvements of this approach, but also of the way and means for disseminating it in the entire organisation, and of the support provided to the designers for using it adequately. The ASN decided to evaluate another topic, which concerns the competencies of operating and maintenance staff in the nuclear power plants. Inspections also cover a large spectrum of relevant human factors topics.

| Seq. No 73 | Country | Article Article 12 | Ref. in National Report 12.2.2, p56 |
|---|---|---|---|

Question/ Comment: This subsection of the report dealing with the Strengthening of Safety Culture at EDF is very short and focuses on the promotion of the six "safety management levers" and some lessons related to some subsequent follow up "capitalization work".

Answer: On the basis of experience feedback from actions carried out since 1997, EDF has put all these actions into a consistent framework of "safety management", consistent with the INSAG and EFQM (European Foundation for Quality Management) documents. Reinforced safety management based on the reference guide produced in this way is a major project for the next three years at EDF.

Read also answer to question No 60:

EDF set up in 1997 an initial safety management outline built on the basis of the use of management tools that involved risk analysis, self-diagnosis, self-evaluation, a sensitive transient state approach, SAREO (Safety Avaibility Radioprotection Environment Observatory), and operational communication. Gradually the sites set up these tools in the department and progress was made on all of them.

During 2004, an extensive job of capitalizing on site methods was carried out; it made it possible to identify efficient methods on the site and showed the need to include the use of all these tools in a more global safety management framework. A "Safety Management Policy Application Guide" offered the opportunity of establishing the link between the safety policy, the Division quality management policy and the use of different tools. It is built on the basis of eight principles of EFQM which themselves structure the Division management policy published in 2004, October.

This is a reference document, presenting for each principle (controlled by results, leadership and constant vision, implication and development of the staff, brushes of apprenticeship and innovation,…) methods confirming the approaches that can be implemented in the field of safety, as well as a limited number of prescriptions. Documents INSAG 4, 13 and 15 and on another level, the quality management approach (EFQM) are the founding documents of this reference document.

This action is not connected with ten yearly outages.

| Seq. No 74 | Country | Article Article 12 | Ref. in National Report p. 58, 12.4.1 |
|---|---|---|---|

Question/ Comment: Concerning human factors aspects in power reactor operation, several examples of inspection principles are mentioned. Is there a guidance how to conduct the inspection according to these principles and how the results can be assessed?

Answer: Some of these topics are supported by an inspection guideline. For example, such guidelines exist for training and qualification, for safety management, for rigor in operation, for control of subcontracted operations.

| Seq. No 75 | Country | Article Article 12 | Ref. in National Report 12.2.2 |
|---|---|---|---|

Question/ Comment: Reference: 12.2.2 Strengthening the safety culture at EDF
In 1997, the director of the Nuclear Operations Department sent a letter to the individual plant managers, entitled "managing nuclear operating safety". As a result of the process set in motion by this letter, each site has developed certain practices. With effect from this date, six "tools"(referred to as "safety management levers") have been promoted and monitored closely, and have also received corporate support: -self-diagnosis, -self-assessment,
Q1: Please explain the specific regulatory methods for the evaluation of safety culture. (Is the safety culture regulated directly? What are the specific items and methods of evaluation?)
Q2: In what manner does the regulator evaluate the self-diagnosis and self-assessment results by the licensee, and use these for regulatory purpose? Are above results reported periodically in the future by the licensee?
Q3: Please explain the specific decision criteria in judging degradation of the safety culture in the events.

Answer: Q1:
ASN assesses safety culture in a qualitative way since quantitative indicators alone are not appropriate. It is more the process of managing the safety and improving it that is examined by the ASN, including the use by the licensee of appropriate levers, tools in this process. The safety culture is considered in the analysis of events as described in the INES scale, throughout inspections dedicated to this topic and also throughout the transparency of the licensee with regard to his relation with the ASN.

For instance, rigor of operating is a priority topic for the program of inspections this year 2005. It focuses on the application of safety culture and safety management policy on the field, day-to-day in the different departments of the NPP. It is about how rigour and vigilance are improved on the site, how requirements are clarified and how that is controlled on the site.

Q2:
The self-diagnosis and self-assessment performed by the licensee are considered of

good quality and they are periodically confronted with ASN's evaluation. But ASN does not want to interfere directly inside the licensee's self assessment process. More generally, what is taken into account by ASN is not directly the results of tools such as self-assessment and self-diagnosis, but the use of these tools in a general approach aiming at improving safety. Thus, a self –assessment in a department or a team may not be efficient if there is not an improvement approach including for instance an analysis of situation, development of actions and measures, etc.

Q3:
Degradation of the safety culture can be measured in different ways such as insights from inspections, analysis of events, etc.

In the events analysis, safety culture is considered as an additional factor, in compliance with the INES scale guide. In particular, ASN takes into account elements raised from the first analysis of the event which may highlight a lack in safety culture, such as:

- violations of operating limits and conditions,
- lacks in quality assurance processes,
- increase of human induced errors,
- insufficient or inappropriate use of feedback experience.

These elements are included in the decision process if they may indicate lacks in management, organisation or attitudes, but they are not included if it is only an individual and specific case.

| Seq. No 76 | Country | Article Article 12 | Ref. in National Report p55 |
|---|---|---|---|

Question/ Comment: Human Performance may be incorporated in the Risk-Informed Implementation Plan.
What kinds of human performance and methodology are required in relation with the implementation of Risk-Informed regulation in your country?

Answer: The ASN do not regulate human performance and methodology in a risk-informed approach as understood with the common meaning of "based on PSA results". However it can be mentioned that human performance is modelled in PSA through methods that were agreed by ASN (MERMOS methodology).

| Seq. No 77 | Country | Article Article 12 | Ref. in National Report p9 |
|---|---|---|---|

Question/ Comment: In 2.3.5(p.9), it is described that "safety management inspection system" was established to get utilities review human and organizational factors.
What are details of the system established? Is this inspection system added newly to existing inspection system of ASN?

Answer: This topic was already covered by inspections since several years, but in 2003 a specific inspection guideline has been elaborated. It concerns mainly issues such as general policy and organisation of the plant for managing safety, resources, staff, organisation and actions of safety quality departments, verification and audits made and corrective actions, etc

| Seq. No 78 | Country | Article Article 12 | Ref. in National Report |
|---|---|---|---|

Question/ Comment

According to estimates from different sources, up to 40% of emergencies/abnormal occurrences at NPPs are caused by human errors (NPP operating, maintenance and servicing personnel). The importance of resolving this problem is stressed in the regulatory documents of IAEA and IEC, which offer various recommendations to reduce the number of events and failures. Therefore it would be desirable to have information on the experience with implementing the recommendations given by IAEA and IEC aimed at reducing the rates of NPP personnel errors leading to emergencies as well as practical substantiation of this kind of experience with statistical data.
What methods of reducing human factor induced failure rates out of those recommended by IAEA and IEC have proved to be most effective (please, give examples from the operating experience and quantitative estimates of results)?

Answer

EDF is developing the acknowledgement of the human factor in operation on the basis of safety management (see answer 60 below). Human Factors specialists have provided the managers with support for their actions. The management principles brought to the foreground: guidance by results, leadership and a constant vision, involvement and development of the staff, apprenticeship and innovation process, are all of a type to add impulse to the progress made.

More often than not, the following methods have been developed on EDF sites: pre-job briefing, risk analysis, hierarchical field visits, use of simulator to prepare for difficult situations, post job briefing. In 2004, human errors causing automatic reactor shutdowns resulted in specific actions on the lowest performing sites, generating overall gains for the nuclear inventory representing approximately 12%. Alignment errors resulting in Events Significant for Safety (ESS), which were the subject of careful managerial monitoring, were also down by almost 50% in 2004.

For the three years to come EDF is going to continue to develop its basic actions (capitalization on experience acquired, development of safety management principles) and push home the guidance of targeted actions concerning theme is identified as representing major vectors for progress.

Read also answer to question No 60:

EDF set up in 1997 an initial safety management outline built on the basis of the use of management tools that involved risk analysis, self-diagnosis, self-evaluation, a sensitive transient state approach, SAREO (Safety Availability Radioprotection Environment Observatory), and operational communication. Gradually the sites set up these tools in the department and progress was made on all of them.

During 2004, an extensive job of capitalizing on site methods was carried out; it made it possible to identify efficient methods on the site and showed the need to include the use of all these tools in a more global safety management framework. A "Safety Management Policy Application Guide" offered the opportunity of establishing the link between the safety policy, the Division quality management policy and the use of different tools. It is built on the basis of eight principles of EFQM which themselves structure the Division management policy published in 2004, October.

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This action is not connected with ten yearly outages.

| Seq. No 79 | Country | Article Article 12 | Ref. in National Report |
|---|---|---|---|

Question/ Comment

Good practice
The ‘Quality order’ and the integration of Human Factors and Quality Management policies are considered good practices.

Are the effects of the integration of Human Factors and Quality Management approaches evident in trending analysis of French operating experience, for example, on the level of safety culture and number of human errors at NPPs?

Answer

The elements added are global for EDF and will be subject to variations according to the sites concerned.
Human errors at the origin of Events Significant for Safety (ESS) are stable for the last three to four years at EDF. This indicator alone is not sufficient for performing trend analysis.

Read also answers to question No 78:
EDF is developing the acknowledgement of the human factor in operation on the basis of safety management (see answer 60 below). Human Factors specialists have provided the managers with support for their actions. The management principles brought to the foreground: guidance by results, leadership and a constant vision, involvement and development of the staff, apprenticeship and innovation process, are all of a type to add impulse to the progress made.
More often than not, the following methods have been developed on EDF sites: pre-job briefing, risk analysis, hierarchical field visits, use of simulator to prepare for difficult situations, post job briefing. In 2004, human errors causing automatic reactor shutdowns resulted in specific actions on the lowest performing sites, generating overall gains for the nuclear inventory representing approximately 12%. Alignment errors resulting in Events Significant for Safety (ESS), the subject of careful managerial monitoring, were also down by almost 50% in 2004.
For the three years to come EDF is going to continue to develop its basic actions (capitalization on experience acquired, development of safety management principles) and push home the guidance of targeted actions concerning theme is identified as representing major vectors for progress.

Read also answer to question No 60:

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This action is not connected with ten yearly outages.

And read also answer to question No 81:
Several factors have been observed and evaluated over more than two years:

- the considerable effect of strengthening the manager leadership and involving the personnel in the results of the teams: the management strengthening factor also applies to the results.
- the dynamic action has led safety to bear on all the results: the best sites in terms of safety are also the best in other fields of performance, thanks to the effect of the management factors developed to improve safety, adding impetus to the other results.
- the development and generalization of high-performance methods and practice recommended by the Nuclear Production Division (management levers).
- the reinforcing of process ruggedness, combined with tighter control of the results and fieldwork among the teams concerning requirements, consulting and checking of activities.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 80 | | Article 12 | page 9 + p. 58 |

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|---|---|
| Question/ Comment | What is the basis of the Safety Management Inspection Programme?<br>How is it carried out?<br>What are the competencies of the inspectors in the area of Human and Organisational Factors? |
| Answer | This topic was covered by inspections since several years, but in 2003 a specific inspection guideline has been elaborated. It concerns mainly issues such as general policy and organisation of the plant for managing safety, resources, staff, organisation and actions of safety quality departments, verification and audits made and corrective actions, etc.<br><br>Inspectors have currently no specific competencies in this field. However, some specific actions such a human factors training programs are on the way in order to improvement this situation. |

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 81 | | Article 12 | sect. 12.2.1 pg. 56 |

**Question/ Comment** The report indicates that for 2002-2005, three avenues of progress have been decided upon for implementing EDF's human factors policy: "improvement of operating means...," "skills management...," and "improvement of day-to-day practices with changes to the organization, as well as individual and collective working methods..." In addition it states that a "safety and human factors" management advisory unit was setup in 2003 as part of the Nuclear Operations Department's senior hierarchy. Please discuss the status of these human factors improvement initiatives, particularly where any results have been observed.

**Answer** Several factors have been observed and evaluated over more than two years by EDF:

- the considerable effect of strengthening the manager leadership and involving the personnel in the results of the teams: the management strengthening factor also applies to the results.
- the dynamic action has led safety to bear on all the results: the best sites in terms of safety are also the best in other fields of performance, thanks to the effect of the management factors developed to improve safety, adding impetus to the other results.
- the development and generalization of high-performance methods and practice recommended by the Nuclear Production Division (management levers)
- the reinforcing of process ruggedness, combined with tighter control of the results and fieldwork among the teams concerning requirements, consulting and checking of activities.

## Art. 13 – Quality assurance

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 82 | | Article 13 | |

Question/ Comment: Is the ISO 9001 Quality Assurance process completed at all units of the Nuclear Operations, Nuclear Engineering and Nuclear Fuels Department?

Answer: ISO 9001 certification is not an EDF Corporate requirement. On nuclear NPP sites, it is evident that certifications are acquired or are in the process of acquisition, more often than not by departments for which the customer-supplier interfaces and relations are important (purchasing, tertiary, nuclear fuel...).

EDF Corporate Units, because of their assignments (in particular with respect to the sites), have all taken the option of acquiring ISO 9001 certification, now acquired by all the Engineering Units and in the process of acquisition by the Nuclear Fuel Division.

What is more, in the field of the environment, EDF and all of its Units are now ISO 14001 certified; it has become a Corporate requirement.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 83 | | Article 13 | p.61 |

Question/ Comment: Reference:13.1 Regulatory requests
Q1: Does the ASN plan to introduce the ISO 9001 or other similar quality management program?

Answer: Yes. ASN is currently organizing its management program according to ISO-9001-2000 standards.
This is in line with the decision of the French State to organizes its accountings with a more indicators-driven spirit, with the new Bill on State Budget for the year 2006. ASN is therefore implementing a group of indicators to insure its effectiveness.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 84 | | Article 13 | |

Question/ Comment: Could you please describe in main lines the regulator´s quality management program : its implementation, certification, applied criteria to be met and evaluation of the quality management.

Answer: ASN is currently organizing its management program according to ISO-9001-2000 standards.
This is in line with the decision of the French State to organizes its accountings with a more indicators-driven spirit, with the new Bill on State Budget for the year 2006. ASN is therefore implementing a group of indicators to insure its effectiveness.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 85 | | Article 13 | Pg 65 Chap 13.4 |

Question/ Comment: In section 13.4 it is stated that the ASN monitors compliance with the Quality Order on the basis of incident feedback and inspection findings on malfunctions.

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| | Does the ASN proactively inspect or audit organisational aspects independently of the utility and industry initiatives (eg WANO) such as: Human performance, Competencies<br>If so, what criteria/guidelines do you apply? |
| Answer | Yes, organisational aspects are considered by the ASN independently of the utility initiatives such as WANO ones.<br><br>Inspections are made on different aspects regarding the compliance with the August 1984 Quality Order. For some of these aspects, ASN developed inspection guidelines for safety management or for rigor in operation in order to address organisational aspects. These guidelines mainly deal with general policy and organisation of the plant for managing safety, resources, staff, organisation and actions of safety quality departments, verification and audits made and corrective actions, etc. |

| Seq. No<br>86 | Country | Article<br>Article 13 | Ref. in National Report<br>page 61f |
|---|---|---|---|

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|---|---|
| Question/ Comment | All the relevant aspects of Quality Assurance and Quality Management are thoroughly discussed, but there is no discussion on the developments of a Quality Management System in the authority. |
| Answer | Same Answer as to Question N° 84:<br>ASN is currently organizing its management program according to ISO-9001-2000 standards.<br><br>This is in line with the decision of the French State to organizes its accountings with a more indicators-driven spirit, with the new Bill on State Budget for the year 2006. ASN is therefore implementing a group of indicators to insure its effectiveness. |

## Art. 14 – Assessment and verification of safety

| Seq. No 87 | Country | Article Article 14 | Ref. in National Report |
|---|---|---|---|
| Question/ Comment | The report describes that after the periodic safety review, the ASN decides on whether or not reactor operations can continue until the next decade outage. On what basis or regulations is this decision-making based? | | |
| Answer | Literally the current regulation (decree 63-1228) provides only for requesting the implementation of measures to insure a safe operation of a BNI (Art. 3) and Periodic Safety Reviews (Art 5). The formal operation licence is given at the commissioning of the plant. However the practice of the ASN is to state about the continuation of operation after each major outages.<br><br>This decision depends on the results of the periodic safety review, that is to say whether non-conformances have been corrected or safety improvements have reached a satisfying level.<br><br>A draft law on transparency and nuclear security and its application decrees are expected to give a sounder legal basis to this process. | | |

| Seq. No 88 | Country | Article Article 14 | Ref. in National Report |
|---|---|---|---|
| Question/ Comment | What were the back-fitting measures at the 900 MWe reactors to be implemented for improving their safety level? What is the time schedule planned for the implementation of these measures? | | |
| Answer | As indicated in the report, each NPP series is submitted every 10 years to a PSR. The 900 MWe series finished this process prior to the second 10-year outage (VD2) in 2003 and began then a new process for the 3rd 10-year outage (VD3). For the 1300 MWe series, the strategic phase associated to the VD2 is completed and the process should be completed in 2005 with the approval of the updated safety analysis report, which was sent in 2004. So, the list of modifications is known for both 900 MWe and 1300 MWe series VD2 PSR, but is under instruction for VD3 900 MWe.<br><br>The implementation of the back fitting measures is as far as possible achieved as a whole batch of modifications during the subsequent ten yearly outage of each reactor of the series.<br><br>Corrective actions may be decided either as a consequence of conformity check or as a consequence of re-evaluation of requirements. Here are some examples of backfitting measures concerning 900 MWe series as a consequence of the VD2 PSR<br><br>- Examples of corrective actions of the first category are given below:<br>- addition of insulation, of complementary heaters, protection of sensors, and definition of procedures to be applied in case of very low temperatures, due to a lower temperature for which safety related systems must be kept operating,<br>- addition of sills and water level sensors in the sumps for internal flooding protection,<br>- implementation of instrumentation to better monitor severe accidents such | | |

as containment pressure and hydrogen measurements

- In the second category, one can find:
  - Automatic make-up device to Reactor Coolant System (RCS) in case of loss of Residual Heat Removal (RHR) system.
  - Automatic interruption of dilution in case of primary pump trip and automatic switch-off suction from Chemical and Volume Control System (CVCS) pumps to Reactor Water Storage Tank (RWST) if the residual power is not sufficient for homogenisation.
  - Automatic isolation of CVCS let down line in case of a total loss of heat sink.
  - Redundant and diversified reactor trip signal to reduce the risk of high-pressure core melt.
  - Several improvements of operating procedures.
  - The programme of surveillance and maintenance of systems.
  - Treatment of possible common mode failures on electrical power distribution.

| Seq. No 89 | Country | Article Article 14 | Ref. in National Report |
|---|---|---|---|

Question/ Comment: What is the scope of the PSA and which operation states are considered?

Answer: The scope of the PSA is the assessment of core damage frequency (PSA level 1) and the characterization of radioactive releases in case of severe accident in terms of nature and frequency (PSA level 2). Both at-power and shutdown states are considered.

| Seq. No 90 | Country | Article Article 14 | Ref. in National Report |
|---|---|---|---|

Question/ Comment: How are the site specific issues covered in the PSA?

Answer: EDF “Reference PSAs” are developed for each of the French NPP series: 900 MWe, 1300 MWe, N4 series. For each series a representative site for off-site power and ultimate heat sink is chosen.

Site-specific issues are not considered in so-called Reference PSAs but in dedicated probabilistic studies (for example assessment of flooding consequences or long-term LOOP).

| Seq. No 91 | Country | Article Article 14 | Ref. in National Report |
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Question/ Comment: In which context(s) is the PSA used to support the Safety Review?

Answer: PSAs are used during the periodic safety review to assess the core damage frequency and its change compared with the assessment made on completion of the previous review, including an analysis of the changes in system characteristics (equipment reliability, for example) and in operating practices.

In addition, identification of the main contributions to the core damage frequency highlights any weak points for which design and operation changes can be studied, or even judged necessary. They can be ranked so as to target the priority work.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 92 | | Article 14 | page 67 |

Question/ Comment What documents the licensee has to submit when applying for licensing of a nuclear facility operation?

Answer As mentioned in the report (§7.2.2.1, p. 26), in France, nuclear installations are currently regulated by the 1963 decree that notably provides for an authorisation decree procedure followed by a series of licences issued at key points in the plant lifetime (see report §18.1.2, p. 107-108) whose content may be specified as follows:

1/ Initial fuel loading in the reactor vessel licence
The first core load can only be delivered to the fuel storage building after authorisation from the Ministers for the Environment and for Industry, granted after examination by the DGSNR :

- of the storage provisions made by the Operator, as presented at least 3 months beforehand ;
- of the conclusions of an inspection carried out shortly before the date set for delivery of the fuel elements.

Moreover, 6 months before fuel loading in the reactor vessel, the operator must send the Ministers for the Environment and for Industry :

- a provisional general operating rules (RGE)
- an onsite emergency plan (PUI) specifying the organisational provisions and measures to be implemented on the site in the event of an accident ;
- a Provisional Safety Analysis Report, which is an update of the Preliminary Safety Report assessed during the authorisation decree application phase, must also be sent.

2/ Initial start-up licences
Afterwards, at least 3 successive licences are required in the startup stages :

- a license for pre-critical hot testing, prior to the first criticality. Those tests are only authorised after issue of the primary system hydrotest certificate in application of a ministerial order ;
- a license for first criticality and power build-up to 90% nominal ;
- a license for power build up to 100% of nominal.

3/ Commissioning license
After this initial start-up and within a time limit stipulated in the authorisation decree (generally 10 years) the operator requests the issue of a commissioning license by the Ministers for the Environment and for Industry.
His request shall be substantiated by :

- the final version of the general operating rules (RGE)
- a revised onsite emergency plan (PUI);
- a Final Safety Analysis Report

Those documents must reflect the experience acquired during the operating period since the initial start-up.

| Seq. No 93 | Country | Article Article 14 | Ref. in National Report page 67 |
|---|---|---|---|

Question/ Comment: Where is the place of the PSA (levels 1 and 2) during the assessment of the nuclear facility safety?

Answer: PSAs are used during the periodic safety review to assess the core damage frequency and its change compared with the assessment made on completion of the previous review, including an analysis of the changes in system characteristics (equipment reliability, for example) and in operating practices.

In addition, identification of the main contributions to the core damage frequency highlights any weak points for which design and operation changes can be studied, or even judged necessary. They can be ranked so as to target the priority work.

During the first step of the periodic safety review (checking that installations are still in conformity with the initial requirements fixed for them), the reference PSA (applying to the whole NPP series) is updated, incorporating the most recent operating experience (identification and frequency of initiating events, equipment reliability data, operating profile), the standard construction condition (design and operation) and new knowledge about the behaviour of the installation obtained from the most recent studies.

Following the periodic safety review, a new version of the reference PSA is produced, taking into account the changes decided on completion of the review process.

| Seq. No 94 | Country | Article Article 14 | Ref. in National Report page 68 |
|---|---|---|---|

Question/ Comment: What is the control by the regulatory body beyond the scope of the Periodic Safety Review, when implementing modifications of systems and/or components, important to safety?

Answer: ASN defined a modifications examination process for safety related materials, which proportions the examination level of the modifications according to their stake for safety. The modifications calling into question the safety demonstration are subjected to the approval of ASN. Within the specific framework of the Periodic Safety Review, ASN examines moreover the relevance of the solutions adopted by the utility, to check that the proposed modifications achieve the aimed safety goal. Moreover, the ASN carries out, within the framework of its inspections, in situ controls of the good realization of the modifications, including requalification tests.

| Seq. No 95 | Country | Article Article 14 | Ref. in National Report 2.3.6, p 10 + p. 73 |
|---|---|---|---|

Question/ Comment: The report seems to state that significant operating experience such as the extreme weather conditions “will be reassessed within the framework of the third ten-yearly outages”. It is noted in subsection 16.2.2.2; page 93, that the DGSNR emergency response centre has been activated on two occasions (28/29 Dec 1999 and 2/3 Dec 2003) due to extreme weather conditions. The final paragraph of section 7.3.2.4.1 (foot of page 34) also seems to indicate that consideration of "significant incidents"

with "nuclear safety implications" is undertaken primarily during the Periodic Safety Reviews (that is, every 10 years).
Please explain whether there would be a potential safety issue related to the delay until a 10-yearly Periodic Safety Review of the assessment of the significance of such major operating experience (which could lead to potentially exposing reactors to an operating period at risk of repeating events experienced elsewhere.).

Answer The word "primarily" used in the report means that backfits derived from operating experience feedback are for the most implemented in the frame of the PSRs, but for significant issues (typically level 2 events), which are only a few ones, the backfits may need to be implemented sooner. In this case, ASN will firstly ask the licensee to implement short-term measures in order to reduce the risks, without waiting the next periodic safety review. As an example, in 2003 in the light of potential impact of failure of recirculation function (filter clogging in the water recirculation sumps), ASN asked EDF to propose measures to remedy the anomaly by the end of the same year. Another example is the Blayais flooding in 1999, which led to corrective measures before the PSRs.

| Seq. No | Country | Article | Ref. in National Report |
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| 96 | | Article 14 | 2.3.8, p11 + p. 74 |

Question/ Comment (See also 14.4.1.3, p74)
The report indicates, on page 11, that "… information collected on recent and future reactors (has) been used to define the orientation of the reassessment associated with the third 10-yearly outages for the 900 MWe series". The report continues to state, on page 74, that "… ASN has defined the orientations for the periodic safety review of the 34 900 MWe reactors in association with their third ten-yearly outage".
Please provide details about the ASN orientations for PSRs associated with the 3rd 10-yearly outages. Please also indicate whether these orientations include requirements/guidance on potential life extension of the 900 MWe beyond their design life of 40 years.

Answer Firstly, the 10 August 1984 order on quality (see report §7.3.1.3, p. 29) provides a general framework for provisions to be taken by any BNI operator to produce, obtain and maintain plant and operating quality standards compatible with safety requirements. This order is applicable to the studies performed in the frame of the periodic safety review.

As mentioned in the report (p. 68) the principles that regulate the conduct of periodic safety review are defined in the 11 December 1963 decree on BNIs (article 5). Then, on a case-by-case basis ASN issues letters to the BNI operators that define the scope of the safety review and specific issues that have to be considered.
For example, for 30-year safety review for 900 MWe, as mentioned in the report (§14.4.1.3, p. 74), the ASN issued a letter in October 2003 instigating the safety review, determining the scope and the limits of the studies to be made by EDF, together with the deadlines to be met to enable the resulting modifications to be integrated on the 900 MWe reactors during their third ten-yearly outages scheduled as of 2008. This letter has been made public on ASN's internet site: www.asn.gouv.fr.

Hereafter are given the headlines of issues to be considered:

1/Internal and external hazards

- o simultaneous failure of equipment non designed to withstand seismic conditions
- o consideration of internal flooding in shutdown states
- o internal explosion
- o fire
- o seismic verification approach
- o Adverse weather conditions
- o Hydrocarbon slick drift on river or seaside

2/Accident studies and radiological consequences

- o cold overpressure
- o Long term phases assumptions for accident studies
- o Steam generator tube rupture
- o Severe accident radiological consequences
- o Containment
- o Beyond design basis equipment
- o Backup of Auxiliary Feedwater System tank
- o Post accident surveillance information

3/Design of systems

- o Design verification of civil engineering structures
- o Functioning of Plant Radiation Monitoring system
- o Reliability of heat removal system of the fuel building
- o Performance of safety injection system
- o Reliability of emergency cooling recirculation function

In terms of ageing, ASN has required EDF to present, for each plant, a file showing that reactors are able to be operated safely after 30 years lifetime. These files will be examined by ASN in the frame of the PSRs. They will include :

- the description of the installation and its operating conditions ;
- the information relating to the manufacture or the realization of the installation being able to contribute to the quantitative analysis of the mechanisms of ageing;
- the assessment of the experience feedback of the behaviour of the installation in service;
- the analysis of obsolescence risks;
- the analysis of components or structures that are replaceable;
- the list of ageing mechanisms.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 97 | | Article 14 | 14.1.2, p67 |

Question/ Comment: It is indicated that “each nuclear power plant is the subject of an average of about twenty inspections a year”.
Please explain whether these twenty inspections form a “baseline” inspection program. Please indicate how many additional inspections are performed following an event or another discovery.

Answer: A minimal periodicity of inspection according to topics of a definite list is required for each BNI and each nuclear site, called “hard core”. According to this hardcore, each nuclear power plant is the subject of an average of about 15 inspections (depending of the number of reactor located on the nuclear site). An average of 3 to 6 additional inspections are performed following an event (depending of the event) and during outages (work site inspections).

In addition it should be recalled that daily contacts are maintained between NPPs and the Nuclear Safety Authority

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 98 | | Article 14 | 14.1.3, p68 |

Question/ Comment: The report appears to put “prime importance” on the periodic safety reviews undertaken every ten years for nuclear facilities. The general process is said (subsection 14.1.3.2) to involve a “two-fold comparison”... “Comparison of the condition of the installations with their design reference …” and “Comparison of the level of installation safety with that required for the most recent reactor”
How does the approach and scope of the periodic safety reviews undertaken by France compare with the Safety Factor approach to Periodic Safety Reviews documented by the IAEA?
Please indicate how long it takes EDF to complete a PSR and how long does it take the ASN to review the results of a PSR.

Answer: Q1/
IAEA safety standard NS-G-2.10 provides a list of 14 safety factors which are taken into account to determine the scope and the list of subjects to be studied during the PSR (each subject can be linked to a safety factor). However, each subject can be approached in terms of conformity (Comparison of the condition of the installations with their design reference) or in terms of safety level.

Q2/

To complete a PSR, a typical duration is about 3 years, from the definition of the scope to the list of the modifications proposed by EDF.

For the regulatory assessment of the PSR of the 900 MWe plants for their 30 years, the ASN review consisted in defining the steps and objectives (1 year), examining the studies performed (1,5 years), and reviewing the conclusions and updated the SAR (1 year).

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 99 | | Article 14 | 14.2.2, p70 |

Question/ Comment: Please elaborate as to how the “EDF safety review” described in this subsection (14.2.2) relates to the Periodic Safety Review and its results (described in 14.1.3). It appears that the “EDF safety review” and the PSR aim at and achieve same or similar objectives and outcome!

Answer: There is no real difference between “EDF safety review” and “Periodic Safety Review” performed by EDF.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 100 | | Article 14 | P67.Ch14.1.2 |

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| Question/ Comment | What kinds of inspection and assessment are performed on the plants in France by external organizations? And how to coordinate all these inspections and assessments? |
| Answer | On average, an OSART is performed each year for a French NPP, as well as three WANO peer reviews every two years. Follow-ups are performed between 12 and 18 months after each of these assessments. Since 2004, to benefit from more frequent international evaluations, EDF- Nuclear Production Division (NPD) has asked WANO for the integration of five foreign peers in the internal assessment teams called Global Safety Assessement (GSA). Three ventures went through in 2004 and three every two years have been programmed for the years to come. |

For the OSARTs, NPD is proposing the name of a site to the Nuclear Safety Authority. In the event of agreement, the latter will send the request to the French government representative at the AIEA. For OSART and all the other external reviews (Peer Reviews) and those internal to EDF, a multi-annual program is updated every year. The alternation between internal and external reviews is taken into consideration by this program.

In addition to these international evaluations, outside the Company, inspections are performed by the Nuclear Safety Authority (unannounced inspections, topic-related inspections, review inspections) for each NPP. These inspections are programmed by the Nuclear Safety Authority independently of the international reviews internal to EDF.

Read also answer to question No 103/2, which describes the internal inspection system of EDF on its nuclear sites:

Global Safety Assessment (GSA)'s offer a means of evaluating the safety, radio protection and environment levels of the various Nuclear Production Division (NPD) entity with regard to prescriptions and ambitions, thus directing and the decisions and actions in such a way as to improve installation safety.

These evaluations are carried out on the basis of safety assessment reference guidelines confirmed by the Nuclear Production Division Management. This reference material is implemented exhaustively, representing more than 400 performance operations assessed by observations in the field, discussions with operational people from all the different professions and examination of the documents. The purpose is to detect in what way the observed methods contribute or do not contribute to achieving safety objectives. The evaluation team includes 14 professional inspectors and a dozen or so pairs of different EDF sites.

Eight evaluation fields are covered: safety management, control, maintenance, transverse support (experience feedback, engineering, modifications, fuel and core physics), radioprotection, environment, fire and states of installations together with an assessment of the Control training service. The field of dismantling is also evaluated when sites have both installations in production and others under dismantling.

Each field is split into themes, themselves split into objectives. Each team is evaluated at several different levels ranging from "excellent" to "unacceptable". This breakdown allows comparison between site's and the monitoring of each site from one evaluation to the following.

These evaluations give the Division and Unit managerial line elements that are likely to help improve safety levels in the form of recommendations, suggestions and good practice methods.

As a complement to the conformity evaluation described previously, the team of auditors will give its perception of the social-organizational aspect of the site, in particular the strong and weak points of the organization, as well as a review of the development risk.

Finally, in some cases, these evaluations may be an opportunity to question once again the pertinence of the prescriptions and ambitions defined by the Nuclear Production Division Management.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 101 | | Article 14 | 14.4.1.1 |

Question/ Comment (Article 14, 14.4,1,1 Conclusion of the 20-year safety review on the 900MW reactors)

It is stated that the wide-range conformity check as part of 20-year safety review for the Fessenheim and Le Bugey reactors led to the detection and correction of non-conformities, in particular concerning the seismic resistance of components.

1. Was the conformity check of the reactors carried out in the safety review based on new geologic and seismologic information and newly-established seismic requirement?

2. What were the method and procedure for the conformity check and what kind of new geologic and seismologic information was taken into account in the conformity check?

3. What were the corrective actions taken on components and structures from the conformity check?

4. What are the other nuclear power plants for which the conformity check was performed and then correction was taken related to the seismic safety?

Answer Q1/

The conformity checks carried out as a part of 20-year safety review for the Fessenheim and Le Bugey reactors included the review of geologic and seismologic information and newly-established seismic requirement. Such new requirements are included in Basic Safety Rule RFS I.2.c, which was published in 1980, i.e. after the plant commissioning. A new analysis is being performed in the same field as part of the 30-year safety review being performed now, due to the replacement of Basic Safety Rule RFS I.2.c by RFS 2001-01.

Q2/

The studies have been performed according to the methodology defined by the Basic Safety Rule, mainly based on the same historical earthquakes than those taken into account for the design of the plant, but better documented due to the use of a national data base named SIRENE

Q3/

There were no corrective actions on the main structures, except part of the internal block walls which were backfitted. Other main corrective actions were:

- reinforcement of non safety classified auxiliary structures to avoid their possible fall on safety classified components
- reinforcement of anchoring of safety classified water tanks

- reinforcement of instrumentation and control relays racks

Q4/
The conformity to design was checked for all the plants of the 900 MWe series. The reinforcements were of the same type, but, as the initial design was made of the basis of a series, it included more margins, which enabled the extension of the reinforcements to be limited compared to Bugey and Fessenheim, which are pre-series plants.

| Seq. No 102 | Country | Article Article 14 | Ref. in National Report A14.4.1.1 P73 |
|---|---|---|---|

Question/ Comment It is mentioned in Article 14.4.1.1 that “probabilistic safety assessments(PSA) were used to highlight failure scenarios, the importance of which had hitherto been underestimated”. France may elaborate whether, PSA is a mandatory requirement of ASN?

Answer The first request for the use of PSA, in order to complement the deterministic analyses and to prioritise safety issues, was issued with ASN letters to EDF in July 1977 and March 1978 respectively related to PWR 900 MWe and PWR 1300 MWe safety options.
Since 1990, PSA results have been extensively and successfully used in France. And in 2002, ASN issued a basic safety rule in order to structure and clarify the use of PSA in the regulatory process. Since then, PSA is a mandatory requirement of ASN. The French basic safety rule (RFS 2002-01) on acceptable methods for PSAs development and applications is available on ASN’s web site: www.asn.gouv.fr.

| Seq. No 103 | Country | Article Article 14 | Ref. in National Report |
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Question/ Comment At the international conference on nuclear installations safety named "Continuous Improvement of Nuclear Safety in a Changing World" organized by IAEA in China (Beijing) and held 18-22 October 2004 the French speaker Mr. H. Robineau mentioned that a "global safety assessment" exists in France, with the help of which at the national level they identify the weakest plant (poor performer) that needs assistance. This global safety assessment is conducted every three years.
1) Why did Article 14 of the National Report fail to mention this kind of assessment?
2) How does this "global safety assessment" look like?

Answer Q1/
Chapter 14 of the reports covers the physical state of the installations and not the internal assessment of the operator, which is covered in chapter 10.2 (see report p. 48) but possibly with insufficient coverage from this point of view. It could be done during the next report. For information, the overall safety review system was presented by EDF during the recent workshop organized by NEA and IAEA in Tokyo at the end of January2005.

Q2/
Global Safety Assessment (GSA)’s offer a means of evaluating the safety, radio protection and environment levels of the various Nuclear Production Division (NPD) entity with regard to prescriptions and ambitions, thus directing and the decisions and actions in such a way as to improve installation safety.

These evaluations are carried out on the basis of safety assessment reference guidelines confirmed by the Nuclear Production Division Management. This reference material is implemented exhaustively, representing more than 400 performance operations assessed by observations in the field, discussions with operational people from all the different professions and examination of the documents. The purpose is to detect in what way the observed methods contribute or do not contribute to achieving safety objectives. The evaluation team includes 14 professional inspectors and a dozen or so pairs of different EDF sites.

Eight evaluation fields are covered: safety management, control, maintenance, transverse support (experience feedback, engineering, modifications, fuel and core physics), radioprotection, environment, fire and states of installations together with an assessment of the Control training service. The field of dismantling is also evaluated when sites have both installations in production and others under dismantling.

Each field is split into themes, themselves split into objectives. Each team is evaluated at several different levels ranging from "excellent" to "unacceptable". This breakdown allows comparison between site's and the monitoring of each site from one evaluation to the following.

These evaluations give the Division and Unit managerial line elements that are likely to help improve safety levels in the form of recommendations, suggestions and good practice methods.

As a complement to the conformity evaluation described previously, the team of auditors will give its perception of the social-organizational aspect of the site, in particular the strong and weak points of the organization, as well as a review of the development risk.

Finally, in some cases, these evaluations may be an opportunity to question once again the pertinence of the prescriptions and ambitions defined by the Nuclear Production Division Management.

Starting in early 2004, the NPD Management asked WANO to have five pairs of foreign people joining the team of EDF inspectors to give an international complementary viewpoint during GSAs.

| Seq. No<br>104 | Country | Article<br>Article 14 | Ref. in National Report |
|---|---|---|---|

Question/ Comment: The Report fails to give information on PSA results for different PWR series. What is the probability of severe core damage and beyond-design release of radioactivity into the atmosphere for French nuclear plants?

Answer: Level 2 PSA on 900 MWe NPP ended in 2004. Its results are under discussion with Nuclear Safety Authority. According to this PSA the frequency of severe core damage and beyond design release of radioactivity into the atmosphere is assessed to be at most equal to $10^{-6}$/year. Beyond design release means release larger than those induced by design basis accident which don't include any core melt or degradation of containment.

| Seq. No<br>105 | Country | Article<br>Article 14 | Ref. in National Report<br>Item 14.1.3 |
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| Question/ Comment | What periodic NPP safety status reports are submitted to the French regulatory body in addition to the safety assessment reports to be presented every 10 years of operation? |
| Answer | Modifications to the safety demonstration have to be stated in the SAR on a yearly basis. The SAR has to be updated every ten years, on the occasion of the PSRs. In addition to periodic safety review reports, the licensee has to keep up-to-date the general operating rules and the on-site emergency plan, modifications of which are submitted to ASN. |

| Seq. No 106 | Country | Article Article 14 | Ref. in National Report |
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| Question/ Comment | Are the results of the probabilistic safety assessment (PSA) taken into account for the periodic safety reviews? Is PSA a part of the safety documentation required to be presented by the licensee to the regulator? |
| Answer | Yes, the results of PSA are taken into account for the periodic safety reviews. For instance, the assessment of the main contributions to the core damage frequency is an element, which can be used to estimate the change in safety level compared with the assessment made after the previous review.<br>Moreover, in the safety analysis report compiled for each periodic safety review, the licensee is required to include a summary of the reference PSA consistent with the reference and operating condition of the reactors. |

| Seq. No 107 | Country | Article Article 14 | Ref. in National Report |
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| Question/ Comment | Are there any plans for lifetime extension of existing NPPs? If yes, are there any regulatory guides or rules ready or under preparation to specify the regulatory requirements to be fulfilled by the licensee asking for the operational extension? |
| Answer | Lifetime is not specified in the French NPP operating licences. However the safe status of an NPP has to be demonstrated at any time (Art. 5 of the 63-1228 decree) and original design studies considered that information of component behaviour after 30 years were needed for the demonstration of their longer term safe behaviour. That is why ASN has required EDF to present, for each plant, a file showing that reactors are able to be operated safely after 30 years lifetime. These files will be examined by ASN in the frame of the PSRs. They will include:<br>- the description of the installation and its operating conditions ;<br>- the information relating to the manufacture or the realization of the installation being able to contribute to the quantitative analysis of the mechanisms of ageing;<br>- the assessment of the experience feedback of the behaviour of the installation in service;<br>- the analysis of obsolescence risks;<br>- the analysis of components or structures that are replaceable;<br>- the list of ageing mechanisms. |

| Seq. No 108 | Country | Article Article 14 | Ref. in National Report |
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| Question/ Comment | Do you apply the same methodology for safety analyses of all PWRs in France? Is the methodology dependent on the type of analysed accidents (e.g., transients, loss of coolant accidents, ATWS)? |
| Answer | Methodologies used for safety analyses are nearly the same for all French PWR series (900 MWe, 1300 MWe and 1400 MWe). The principles are the same for all series. Nevertheless, sometimes, the analysis rules can be slightly different, especially for the last designed series (1400 MWe).<br><br>On the other hand, the methodology strongly depends on the type of analysed accidents. Particularly :<br>- the design accidents (e.g. loss of coolant accidents) are studied on a deterministic way, with conservative assumptions and rules (e.g. allowance for systematic margins, implementation of the single failure criterion, allowance for safety grade equipment alone...)<br>- the complementary operating conditions (e.g. ATWS) are studied on a different way, using a probabilistic approach. |

| Seq. No 109 | Country | Article Article 14 | Ref. in National Report |
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| Question/ Comment | Does the regulator accept best-estimate approach with uncertainties for licensing safety analyses or is it limited to conservative deterministic approach? |
| Answer | In the licensing process of French NPPs, the safety analyses rely essentially on deterministic approach based on the concept of defence in depth and probabilistic approaches are used to supplement the conventional deterministic analyses.<br><br>In probabilistic analyses, best-estimate approach is clearly taken into account since PSA have to be as realistic as possible. Some deterministic analyses such as the calculation of the public exposure dose may use best estimate approach but the majority is base on conservative approach. |

| Seq. No 110 | Country | Article Article 14 | Ref. in National Report |
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| Question/ Comment | Are there investigations (research) made on water chemistry in the sumps after LOCA in French NPPs with PWRs? What is the pH of water accumulated in the sumps? Is (is not) there any chemical reaction between sump water and fibres of pipeline insulation material or any other construction materials encompassed in the containment? |
| Answer | In the event of accident by rupture of the primary circuit, the debris generated at the breach are subjected to a sprinkling of water comporting the average characteristics equal to pH of 9,4 (presence of sodium hydroxide NaOH) and boron concentration equal to 2500ppm.The materials present in the building engine are qualified to satisfy with the functional requirements under such conditions.<br><br>Concerning the risk of clogging the sumps filters in case of an accident, the current knowledge do not make it possible to apprehend specifically the combined effects of the various parameters likely to generate chemical reactions between the various involved compounds. The influence of the sumps water temperature and its evolution during the accident, are still in debate. Research tasks with international cooperation are in progress on the subject. |

| Seq. No 111 | Country | Article Article 14 | Ref. in National Report |
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Question/ Comment: What is the time limit for the fire resistance of fire protection doors in French NPPs? Is it standardised for all NPPs operated in France?

Answer: The time limit for fire resistance of fire protection doors within French NPPs is :

- for safety compartments 90 minutes based on qualification process (ISO 834),
- for other cases (security, investment protection, etc....) the duration of fire resistance depends on the overall fire load contained in the related compartment (from 30' to 90').

These rules are entirely standardized for all French NPPs.

| Seq. No 112 | Country | Article Article 14 | Ref. in National Report |
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Question/ Comment: What is the actual status of PSA preparation and use for NPPs. Do you perform the PSA level-1, level-2 or even level-3? Do you use integrated PSA models (full power and shut down) to evaluate the safety of French NPPs? Is the risk profile of French NPPs balanced? What are the dominant contributors to the risk (PSA level-1 and PSA level-2) in French PWRs?

Answer: PSAs are used in France since 20 years to supplement the conventional deterministic analyses. They are considered by ASN as an interesting tool in the definition and prioritisation of the actions to be taken in order to attain or maintain a satisfactory safety level. Their main applications for French NPPs include the following safety areas:

- Periodic safety review,
- Probabilistic event analysis,
- Design of future reactors,
- Importance of systems and equipment with regard to safety,
- Operational technical specifications.

Since 1990, a level-1 PSA has been developed in France which covers now all internally initiated events except aggressions, including all applicable reactor states including shutdown. In 2004, the scope of level-1 PSA performed by IRSN for 900 MW plants was extended to include an internal aggression such as fire. In the same year, as mentioned above, the utility developed a level-2 PSA for 900 MW plants covering all applicable reactor states including shutdown.

PSA level 1 have been developed for all EDF NPP series and for the French EPR Project for power and shutdown states. A level 2 PSA has been achieved for EDF 900 MW series and is in preparation for the 4-loop EDF NPP (1300 and N4). A level 2 PSA is foreseen for the French EPR Project.
Integrated PSA models are used to evaluate the safety of French NPP during periodic Safety Review.

The risk profile is balanced. The main contributors for level 1 PSA are the loss of 6,6 kV safeguard switch board by common cause failure and failures of reactor cooling pump and transients without reactor trip caused by rod blockage.

The main contributors for level 2 PSA are heterogonous dilutions with large early releases and situation with basemat melt through with late releases.

At the moment, it is not scheduled to develop a level-3 PSA.

| Seq. No 113 | Country | Article Article 14 | Ref. in National Report |
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Question/ Comment: Do the SARs include safety analyses for low power and shut down plant operating modes or the safety analyses in the SARs limited to full power only?

Answer: The initial conditions considered for design basis accidents discussed in the SAR cover all plant operating conditions, from full power operation to cold shutdown, including low power operation.

| Seq. No 114 | Country | Article Article 14 | Ref. in National Report |
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Question/ Comment: Are the beyond design basis accident (severe accident) analyses included in the licensing process of French NPPs? How do you select the list of analysed scenarios?

Answer: Although severe accident were not taken into account at the initial PWR design stage, they are the subject of specific provisions to limit their consequences for the environment and the public. These provisions are of technical nature (containment venting system, passive autocatalytic hydrogen recombiners, etc.), of documentary nature (Severe Accident Management Guide) and organisational nature (the management of severe accident is explicitly provided for in on-site and off-site emergency plans).
Thus, “ultimate” beyond design procedures "U2" (containment leakage), "U4" (base mat erosion) and "U5" (containment venting through a sand bed filter) are described in the Safety Analysis Report, which is a licensing document.

| Seq. No 115 | Country | Article Article 14 | Ref. in National Report section 14.2.2, p 71 |
|---|---|---|---|

Question/ Comment: It is stated that the programme for additional investigation comprises non-destructive checks that are spread over several units and carried out on the occasion of the ten-yearly outages. The aim is to confirm the validity of the scenarios (degradation modes) on which the basic preventive maintenance programmes are based. The programme is implemented at the start of the ten-yearly period.
Since the programme for additional non-destructive check investigation is spread over several units, does it means that each unit comprise only designated part of that program, so the programme is covered exactly 100 % by all units involved? Please, state what exactly is included in that programme (e.g. ISI, IST…?)!

Answer: The basic preventive maintenance programs (PBMP) and the special maintenance programs in effect provide for an exhaustive list of inspections during the operation of the equipment, according to the risks of confirmed or potential degradation or as defence-in-depth steps.

To strengthen even more defence-in-depth, these programs are completed by a Complementary Investigation Program (PIC) applicable to areas considered not to be sensitive; this is to confirm operator's hypotheses about the absence of any major degradation occurring in service in the areas not covered by PBMP programs or special maintenance programs.

A PIC applies to a plant unit (900 MWe, 1300 MWe) and is implemented on the first unit or on many units of each series according to the analysis, (900 MWe, 1300 MWe,…). Essentially it consists of non-destructive examination but if necessary, may include component samples taken for laboratory expertise. Each examination or sample applies to one of several plant units and all the results obtained correspond to 100% of the PIC. The choice of plant unit or units is made according to special events or specific characteristics regarding such and such a plant unit, which may represent an aggravating factor for the suspected type of degradation.

The first PIC was implemented during the second 10-year inspections of the 900 MWe plant units: it covered the Main Primary Circuit (CPP), the Main Secondary Circuit (CSP) and the pipes and tanks of the RRA (Residual Heat Removal System), ASG (Auxiliary Feedwater System), RIS (Safety Injection System) and RRI (Component Cooling System) systems: 2384 checks were carried out on the sites and a dozen or so components were removed and evaluated in the laboratory. The only degradation in service was found to be a RRI negative pressure generating device.

A second PIC was defined to the second 10-year inspections of the 1300 MWe plant units. It will be implemented in 2005 and cover the CPP, the CSP and the pipes and tanks of the RRA, ASG, RIS, PTR (Reactor Cavity and Spent Fuel Pit Cooling ant Treatment System) and EAS (Containment Spray System) systems.

A third PIC is currently being defined for the third 10-year inspections of the 900 MWe plant units. It will be implemented in 2009 and cover the CPP, the CSP and the pipes and tanks of the RRA, ASG, RIS, RRI, PTR, EAS and RCV (Chemical and Volume Control System) systems. In addition, it will be extended to electrical equipment, instrumentation and control-test equipment and to containment enclosures and other civil engineering structures.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 116 | | Article 14 | p. 67 |

Question/ Comment

There is a lot of emphasis on risk evaluation/risk based approached in licensing process based on PSA level 1 and 2 studies in other countries reports, however there is very little information on PSA studies and PSA application in the French report. We are aware that EdF and the ASN (IRSN) developed two independent PSA Level 1 and 2 studies for all different types (series) of French plants. In section 19.2.7 of the report evaluation of significant event using PSA is briefly described – is this the only application of PSA in France? Are there any plans to use PSA or “risk based” approach in other application?

After “September 11” events in the USA many utilities performed re-evaluation of plant structures and building to reinforce and verify the bases for aeroplane crash type of accident. Were similar analyses carried out in France by EdF?

Answer

Q1/
Since the end of year 2001, the elaboration and the use of PSA studies is controlled

in France by a Basic Safety Rule (in French Règle Fondamentale de Sureté – RFS) issued by the Nuclear Safety Authority. In this RFS, different uses of PSA studies are foreseen, namely:

- safety re-evaluation. It has been the case for the 900MWe plants recently, in view of their 3rd ten yearly re-evaluation. PSA level 1 and level 2 have been issued, in order to define the outlines of necessary back fittings. A tentative PSA on internal hazards (fire) has also been under examination.
- significant events examination, which is quoted in the question. A yearly balance of events is performed.
- future reactor design. PSA (level 1, level 2, hazards) is of large use for EPR design and evaluation, especially for RRC conditions (Risk Reduction Category of events, including "beyond design" conditions and severe accident conditions),
- systems and equipment important for safety : PSA techniques are useful to identify such systems and equipment, as regards Tech Specs, periodic tests and maintenance programs. EDF has implemented an important optimisation program of Reliability Centered Maintenance (in French Optimisation de la Maintenance par la Fiabilité – OMF) on all PWR series.
- Operation Technical Specifications design. PSA studies are able to highlight the best shutdown state combined with the best way to reach it. Such studies are used by EDF for permanent specs justification as well as case-by-case waivers, e.g. on grid connections.

Plants used PSA mainly to support day-by-day demands to the French Nuclear Safety Authority.

Q2/
Analyses of airplane crash resistance have been performed for French NPPs. This issue is addressed in specific studies, which are also related to security and physical protection measures.
By nature the security measures are not included in the scope of this Convention and, in compliance to French law, any information related to such measures cannot be disseminated to the outside without a confidential agreement between two Governments. Such agreements exist between France and some Contracting Parties, to which it should be referred for further information on the topic.

| Seq. No 117 | Country | Article Article 14 | Ref. in National Report 14.4.1.1,3rd.paragra |
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Question/ Comment: This paragraph indicates that it was considered necessary to incorporate modifications in several systems to improve their reliability, among them the auxiliary steam generator feedwater system. It would be advisable to include a brief description of the design modification incorporated in this system (and the underlying reasons for it).

Answer: The problem here referred to takes its origin in the extended inoperability of one channel of the 6.6KV backed-up electrical system (LHB) in unit 4 at CRUAS NPP (following a fire on October 30, 1990) led to an assessment of the 6.6KV backed-up electrical system ( LH = LHA+LHB) common-mode failure occurrence frequencies and of the related core meltdown risk. In view of the significant risk identified by this study, EDF undertook to increase the reliability of the control provisions available in the event of a common-mode LH failure to minimize the core meltdown hazard.

For example, the reliability of the steam generator makeup water supply system was enhanced by allowing the use of the condensate extraction system (CEX) pumps as a backup for the ASG turbine-driven pump in the Emergency Response Team Guide in the case of an LHA-LHB failure. The electrical modifications are designed to maintain all the feedwater-related equipment items necessary for emergency ASG (emergency feedwater system) operation, notably the extraction pumps and their supporting functions, and control and instrumentation of the ARE (normal feedwater system) feedwater low-flow control system in the case of an LHA-LHB failure. These modifications involve creating a new power supply system for selected electrical switchboards and channel A protection units from the permanent auxiliary switchboards. On the CP2 series (900 MWe), a direct link between the extraction pump discharge line was also created.

In addition, the reliability of the primary pump seal injection function was improved to face the failure of the turbine alternator (LLS system), which supply, in this case, the injection pump RIS 011 PO with electricity. An automatic switching of the pump RIS 011 PO power supply on the common LKI switchboard (380V) was added.

| Seq. No 118 | Country | Article Article 14 | Ref. in National Report page 68 |
|---|---|---|---|

Question/ Comment: What is referred to by the term “realistic risk reduction approach”?

Answer: ASN’s policy is to continuously improve safety and not only to maintain it. Periodic safety reviews aim at checking that the BNIs are still in conformity with the initial requirements fixed for them, and also improving their level of safety, taking into account experience feedback and state of the art. Realistic means that these improvements are required as far as they are economically and technically achievable. The need for improvement is discussed on a case-by-case basis.

| Seq. No 119 | Country | Article Article 14 | Ref. in National Report page 72 |
|---|---|---|---|

Question/ Comment: In case EDF makes use of Probabilistic Safety Assessments (PSAs) in the assessment of the reference system, how are the risk analyses reviewed under the authority of the regulatory body and, in particular, how does the regulatory body make sure that the risk analyses well reflect the current configuration of the reference system?

Answer: In 2002, ASN issued a Basic safety rule (RFS 2002-01) on acceptable methods for PSAs development and applications. The rule says, inter alia, that the licensee has to perform a reference PSA consistent with reference and operating conditions. The reference PSA is initially examined at each safety review and on this occasion, the regulatory body makes sure that it meets the requirement of the basic safety rule. Then, whenever the licensee makes use of PSAs, he has to justify his results so that it is always possible for the regulatory body to make sure that the risk analyses well reflect the current configuration of the plant safety reference system. One has to note that French PSA are very detailed and well validated since ASN’s technical support, IRSN, performs its own PSAs and compare them to EDF's ones.

| Seq. No 120 | Country | Article<br>Article 14 | Ref. in National Report |
|---|---|---|---|
| Question/ Comment | 1. What is the realistic approach to risk minimisation as applied for nuclear installation safety enhancement ? | | |
| Answer | Same answer as to question No 118:<br><br>ASN's policy is to continuously improve safety and not only to maintain it. Periodic safety reviews aim at checking that the BNIs are still in conformity with the initial requirements fixed for them, and also improving their level of safety, taking into account experience feedback and state of the art. Realistic means that these improvements are required as far as they are economically and technically achievable. The need for improvement is discussed on a case-by-case basis. | | |

| Seq. No 121 | Country | Article<br>Article 14 | Ref. in National Report |
|---|---|---|---|
| Question/ Comment | 2. How the risk is defined and measured? | | |
| Answer | The risk can be defined by the couple probability versus consequence. ASN considers that deterministic and probabilistic approaches are both relevant to address the risk. Although PSAs are a useful tool in order to quantitatively measure the risk, there are some aspects such as the safety culture that cannot be measured in a quantitative manner. For instance, the Dampierre NPP was put under a reinforced surveillance in 2002 on the basis of a qualitative judgement made by ASN. | | |

| Seq. No 122 | Country | Article<br>Article 14.1 | Ref. in National Report |
|---|---|---|---|
| Question/ Comment | There are several different plant generations in use, each with their own safety cases, which respectively are mutually different from each other. How does France ensure that the earliest plants remain acceptably safe when the safety requirements for modern plants keep evolving into more and more stringent direction? | | |
| Answer | This issue is addressed through the periodic safety review. Oldest plants are compared to the newest design, including future plants such EPR, in order to see which improvements from new design can be implemented on earlier designs.<br><br>Read also answer to question 36 and to question 96:<br>Firstly, the 10 August 1984 order on quality (see report §7.3.1.3, p. 29) provides a general framework for provisions to be taken by any BNI operator to produce, obtain and maintain plant and operating quality standards compatible with safety requirements. This order is applicable to the studies performed in the frame of the periodic safety review.<br><br>As mentioned in the report (p. 68) the principles that regulate the conduct of periodic safety review are defined in the 11 December 1963 decree on BNIs (article 5). Then, on a case-by-case basis ASN issues letters to the BNI operators that define the scope of the safety review and specific issues that have to be | | |

considered.

For example, for 30-year safety review for 900 MWe, as mentioned in the report (§14.4.1.3, p. 74), the ASN issued a letter in October 2003 instigating the safety review, determining the scope and the limits of the studies to be made by EDF, together with the deadlines to be met to enable the resulting modifications to be integrated on the 900 MWe reactors during their third ten-yearly outages scheduled as of 2008. This letter has been made public on ASN's internet site.

Hereafter are given the headlines of issues to be considered:

1/Internal and external hazards

- simultaneous failure of equipment non designed to withstand seismic conditions
- consideration of internal flooding in shutdown states
- internal explosion
- fire
- seismic verification approach
- Adverse weather conditions
- Hydrocarbon slick drift on river or seaside

2/Accident studies and radiological consequences

- cold overpressure
- Long term phases assumptions for accident studies
- Steam generator tube rupture
- Severe accident radiological consequences
- Containment
- Beyond design basis equipment
- Backup of Auxiliary Feedwater System tank
- Post accident surveillance information

3/Design of systems

- Design verification of civil engineering structures
- Functioning of Plant Radiation Monitoring system
- Reliability of heat removal system of the fuel building
- Performance of safety injection system
- Reliability of emergency cooling recirculation function

| Seq. No 123 | Country | Article Article 14.1 | Ref. in National Report |
|---|---|---|---|

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| Question/ Comment | When a safety problem is identified in series-produced units where it cannot be fixed quickly for all units, is it the French practice to implement interim corrective measures while the fixes are being developed and implemented? As an example, what interim measures are in place to cope with the "sump clogging" problem generic to PWRs, to cover the many years needed to implement of the currently envisaged corrective measures? |
| Answer | When a safety problem is identified in series-produced units where it cannot be fixed quickly for all units, the French practice is to examine the need and the feasibility to implement interim corrective measures while the fixes are being developed and implemented. This need is evaluated considering a seriousness scale which takes into account the probability of the situation to be corrected and its possible consequences. |

As an example, interim measures were recently adopted by EDF to reduce the risks associated with the "sump clogging" problem generic to PWRs, during the time needed to implement the final corrective measures.

The durations allowed by ASN for treatment of non-conformances are balanced according to the impact on the installations safety, to the probability of occurrence of the initiator likely to generate the defect, and to the feasibility of repairs. The implementation of palliative provisions, in material term (temporary modification), or organisational (control, maintenance, surveillance) may allow to accept longer times.
In the case of the anomaly relating to the risk of the sumps filters clogging in accidental conditions, the transitional provisions retained by the utility EDF concern:

- Opening of the pre-filters doors of the 900 MWe plant units;
- Avoidance of powdery heat insulation installation (microtherm) in the Building Reactor;
- Possibility of supplying in water the tank of the reactor cavity and spent fuel pit cooling and treatment system.

With respect to the operation of the Safety Injection System (RIS)/ Containment Spray System (EAS) with sump in recirculation mode, a number of measures have been taken or analyzed to minimize the generation of debris or the consequences of sump fouling and its impact on the recirculation function. A possible answer could be:

Opening of the prefilter doors of the 900 MWe plant unit.
In this plant unit, the RIS/EAS sumps are equipped with large mesh pre-filters a few meters upstream. In the case of these filters becoming fouled, in certain circumstances, a damming phenomenon may occur, resulting in the filters becoming clear. In this event, there is a risk of air being drawn in at the Safety Injection System and Containment Spray System pumps. Accordingly, a provisional arrangement (EDF-DPN-DT n°192) was issued in early 2004 in order to have the prefilter doors open and eliminate any risks of dams forming.

Avoidance of powdery heat insulation installation (microtherm).
Powdery heat insulation debris considerably increases the load losses from filter debris mattresses. EDF took the decision in early 2004, insofar as possible, not to install this type of heat insulation inside the Reactor building during operational or maintenance work.
Further, a decision was reached at the same time to minimize the quantity of this type of heat insulation during the future replacement of the SGs. Accordingly, this provision, applied to the replacement of the SGs in plant unit 4 at Tricastin has made it possible to reduce the amount of powdery heat insulation by a factor of 20 or so compared to the initial predictions.

Optimization of control procedures.
The control procedures were analyzed to identify any potential optimization sources with respect to fouling risks. For the "short term" aspect before the fitting out of the emergency teams, the current procedures already appear to be satisfactory and it will be necessary to ensure that the change of control does not degrade the overall safety level of the installations. It is also noteworthy that the

limitation of the Safety Injection System and Containement Spray System flow-rates already appears in the procedures (shutdown of a Containment Spray System line, shutdown of Safety Injection System pumps or realignment…) when so permitted by the parameters. Therefore, it is unadvisable to go any further in the short term flow rate reduction.

Reactor Building cleanliness.
On the basis of the survey of the site methods used, it has been verified that the reactor building cleanliness was satisfactory. The data obtained from this survey was added to the reference material for the purpose of sizing hypotheses.

Other avenues have been explored, for instance the design of deflectors to trap debris upstream of the filters, but in the same way as for the control procedures, these avenues were not deemed to be pertinent and did not lead to any concrete actions.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 124 | | Article 14.1 | p. 67, 14.1.2 & 14.1 |

Question/ Comment: Is the analysis of low-power and shutdown states included in the PSR?

Answer: The analysis of low power and shutdown states is included in the PSR since their importance was discovered in France at the end of the 1980s.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 125 | | Article 14.1 | p. 68, 14.1.3.2 |

Question/ Comment: What is the reference for the comparison of the level of installation safety? Is it the EPR or the N4 plants?

Answer: Up to the 20-year safety review for 900 MWe and 1300 MWe reactors, the reference for the comparison of the level of safety was N4 series. From the 30-year safety review for the 900 MWe reactors, the reference is EPR.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 126 | | Article 14.1 | 1st para.,14.1.3.2, |

Question/ Comment: Reference : 14.1.3.2 , page68
It seems a good practice as for safety reassessment.
Q1: Regarding best international practices, how do you find them? Are there any intended country practices for comparison?

Answer: ASN is involved in international relations with its counterparts in many countries having nuclear installations all over the world as well in IAEA or NEA safety activities. Thus, ASN is aware of various international practices that can be considered by its Advisory committee in order to find the best ones.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 127 | | Article 14.1 | 14.2.2,p72 |

Question/ Comment: Reference: 14.2.2 ”Assessment of the reference system by EDF
”...the most sensitive issues are assessed with regard to their impact on the level of safety of the reactor. When it is apparent that they are sufficiently important and

that this importance far outweighs any other disadvantages there may be, the safety requirement reference system is modified. If necessary, verification studies are carried out again..."

Q1: It is understood that the most sensitive issues are determined by probabilistic safety assessments to determine the priority of components form viewpoint of safety classification. Please add more detailed contents of the most sensitive issues.
Q2: What kinds of criteria are applied to judge whether the safety requirement reference system is modified? Do they use the judgment criteria such as guideline instructed by nuclear safety regulatory authority? If so, please describe additional information on the judgment criteria, decision making procedures and the person who has responsibility for the decision making.
Q3: In the same paragraph, it is described that if necessary, verification studies are carried out again. Do they carry out the verification studies on the following topics;
- Integrity evaluation for neutron irradiation embrittlement of reactor pressure vessel
- Stress corrosion cracking growth for nickel base alloys of reactor pressure vessel penetrations
- Non-destructive Examination for detestability and sizing of stress corrosion cracking of the nickel base alloys.
If so, please add the outlines of these verification studies.

Answer Q1/

Some of the most sensitive issues are determined using PSA. It is the case for issues related to the behaviour of systems to mitigate an initiating event. Here are some example of improvements which come from PSA:

- Automatic make-up device to Reactor Coolant System (RCS) in case of loss of Residual Heat Removal (RHR) system.
- Automatic interruption of dilution in case of primary pump trip and automatic switch-off suction from Chemical and Volume Control System (CVCS) pumps to Reactor Water Storage Tank (RWST) if the residual power is not sufficient for homogenisation.
- Automatic isolation of CVCS let down line in case of a total loss of heat sink.
- Redundant and diversified reactor trip signal to reduce the risk of high pressure core melt.
- Several improvements of operating procedures.

However, it is not the only way for detecting important issues. Others are mainly:

- operating feedback, for instance Blayais incident of external flooding
- evolution of rules from the authority, for instance those applicable to seismic hazard
- comparison to newer standards

Q2/

To judge whether or not the safety requirement reference system has to be modified, all the consequences of the proposed change are evaluated according several aspects including safety, radiological or environmental aspects, but also costs, availability,… This cost to benefit approach enables the proposed changes to be priorized. This hierarchy proposed by EDF may be modified after assessment by

Nuclear Safety Authority and IRSN, and discussion in Advisory Committee of experts for nuclear reactors.

Q3/
The integrity of the pressure vessel due to neutron irradiation embrittlement is evaluated all along the lifetime of the plant. The design value of the fluence is periodically re-evaluated in the framework of the normal monitoring of the plant. The evaluation of the integrity of reactor pressure vessel penetrations is examined in the same framework as all the other nickel base alloys zones of the primary components: steam generator divider plate, core support lugs, reactor pressure vessel nozzles. Independently of a safety review, a specific program based non-destructive investigations is performed. An analysis has been made to determine the most critical zones (in the sense of most sensitive zones to stress corrosion cracking), to be investigated at first.

| Seq. No 128 | Country | Article Article 14.1 | Ref. in National Report 14.4.1.2 |
|---|---|---|---|

Question/ Comment: Reference: 14.4.1.2 The 20-year safety review for the 1300 MW reactors “...In 2002, the ASN consulted its Advisory Committee for nuclear reactors concerning the validity of the engineering studies conducted by EDF., and considered that some proposals could not be accepted as they stood…”

Q1: What are the EDF’s proposals that could not be accepted by the ASN? What are the reasons why the ASN made such judgments for these proposals? Are the judgments for the proposals applied also to other series of the nuclear power plants?

Answer: In 2002, ASN consulted its Advisory Committee for nuclear reactors concerning the validity of the engineering studies conducted by EDF, and considered for instance that utility’s proposal for an evolution of calculation methods of the radiological consequences of design basis accidents aiming at using more realistic methods and assumptions, was not acceptable as they stood. Indeed, ASN considered that the same studies rules were to be taken for the evaluation of the radiological consequences of the accidents and for the design basis accidents studies presented in the safety analysis report. Thus, the calculation of the released activity was to take into account conservative assumptions and methods. On the other hand, the evaluation of the radiological impact on the human and the environment (dispersion of activity released, transfer of the radioactive products in the food chain, doses calculation) was to be carried out with assumptions and methods known as "realistic".

ASN made this judgement in the framework of the 20-year safety review for the 1300 MW reactors and that therefore applied firstly to this series of reactors. A new dossier will be submitted by EdF for assessment to the Advisory Committee for nuclear reactors in 2005.

| Seq. No 129 | Country | Article Article 14.1 | Ref. in National Report A14.1.3.2 P69 |
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Question/ Comment: It is written that “after the periodic safety review, the ASN decides on whether or not reactor operation can continue until the next ten-yearly outage”. France may

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| | elaborate whether the decision regarding reactor operation on the basis of satisfactory 10-yearly periodic safety review is also valid for reactor operation beyond design life or there are additional regulatory requirements for allowing operation? |
| Answer | There is no specified lifetime in the authorizations given by ASN. The possibility to carry on operation is stated by ASN, after examination of a justification file provided by EDF. Obviously, the justification gets more and more difficult when the plants gets older.<br><br>Read also answer to question No 87:<br>Literally the current regulation (decree 63-1228) provides only for requesting the implementation of measures to insure a safe operation of a BNI (Art. 3) and Periodic Safety Reviews (Art 5). The formal operation licence is given at the commissioning of the plant. However the practice of the ASN is to state about the continuation of operation after each major outages.<br><br>This decision depends on the results of the periodic safety review, that is to say whether non-conformances have been corrected or safety improvements have reached a satisfying level.<br>A draft law on transparency and nuclear security and its application decrees are expected to give a sounder legal basis to this process. |

| Seq. No 130 | Country | Article<br>Article 14.1 | Ref. in National Report |
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| Question/ Comment | Since ASN relies on the expertise of outside technical organizations, could you mention the process of design changes approval and who is in charged with control of modification? |
| Answer | Modification to the operating documents (general operating rules, on-site emergency plan) is submitted to ASN's approval. Modification of equipment important to safety is submitted to ASN's approval depending on the safety significance of the modification. Decision is taken by ASN, after an examination that is generally performed by its technical support organisation the IRSN, on request of ASN. |

| Seq. No 131 | Country | Article<br>Article 14.1 | Ref. in National Report<br>14.1.2,page 67 |
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| Question/ Comment | This paragraph indicates that on average some twenty inspections are performed at each plant each year, without including the technical meetings between the operators and the regulator. Do these inspectors belong to the ASN? Do they have any support from technical specialists in the areas inspected? What issues are included in these Inspections? Is there any pre-defined systematic approach to identify the areas to be inspected each year? |
| Answer | The inspectors belong to ASN and they may be supported by IRSN specialists. (See Report § 1.2.1.1 page 116).<br><br>Read also the answer to part 1 of question No 35:<br><br>A minimal periodicity of inspection according to topics of a definite list is required |

for each BNI and each nuclear site, called “hard core”. An annual inspection programme is determined by the ASN. It takes into account, inspections already carried out, DRIRE and ASN information on various plants and progress made on technical subjects under discussion between the ASN and the operators. It is prepared using a methodical approach defining the hard core, annual priority national topics and suitable coverage of the different sites.

This programme is not communicated to BNI operators. It includes BNI’s name, inspectors’ name, topic and announced or unannounced characteristic. Some topics may preferably be the subject of unannounced inspection, such as work site inspections during outages, solid or liquid waste, fire protection, radiation protection, and emergency preparedness. Other topics need licensee specialists to be present: such inspections are announced.

There is no legal basis defining the ratio between announced and unannounced inspection. The ratio generally lies between 15 and 25%.

| Seq. No | Country | Article | Ref. in National Report |
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| 132 | | Article 14.1 | 14.1.3.2,p.68,last p |

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| Question/ Comment | This chapter 14.1.3.2 generally describes the scope of the periodic safety reviews. Firstly comparing the status of the installation with its initial design (reference design), taking into account the changes incorporated since construction (conformity check), and secondly examining the level of safety of the installation as a result of application of the most recent safety requirements made of the most modern reactors, comparing them to the best international practices and the lessons learned from operation of the installation. How are these international practices selected? What method is applied to require application of the new standards? Is this process performed for each unit, for each plant or for each “generation” of reactors? |
| Answer | The periodic safety review is performed both for each plant with regard to external hazards that are site specific, and for each series of reactors for generic studies that are described in the corresponding "standard" safety analysis report. Best practices are derived from IAEA safety standards and from discussion between ASN and its foreign counterparts (bilateral or through IAEA, NEA, etc. meetings).<br><br>ASN applies a realistic risk reduction approach when requiring the application of new requirements on existing reactors, issued after the assessment by its Advisory Committee. That is to say, the improvements in the safety level of existing reactors are required as far as they are economically and technically achievable. |

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 133 | | Article 14.1 | 14.1.3.3,page 69 |

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| Question/ Comment | This chapter 14.1.3.3 indicates that the Order of 10th November 1999 requires a complete inspection of the primary and secondary circuits to be performed every ten years, followed by a hydrostatic test. At what pressure is this hydrostatic test carried out, in relation to the design pressure of the system? Is there any possibility of the licensee achieving exemption from this test by adequate justification? |
| Answer | This hydrostatic test is carried out at 1.2 times the design pressure of the system:<br>- 206 bar (20.6 MPa) for the main primary circuit of any PWR |

- 90 bar ( 9.0 MPa) for the 900 MWe PWR main secondary circuit
- 106 bar (10.6 MPa) for the 1300 MWe PWR main secondary circuit
- 108 bar (10.8 MPa) for the 1450 MWe PWR main secondary circuit

There is no possibility for the licensee to be exempted from this test, even by adequate justification.

| Seq. No 134 | Country | Article Article 14.1 | Ref. in National Report Sect. 14.1.3 |
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**Question/ Comment** The report describes the general process for the periodic safety reviews (conducted every 10 years during the 10-yearly outages). It is mentioned that the ASN reviews the results of these periodic safety reviews and decides whether or not reactor operations can continue for until the next 10-yearly outage. Please discuss some of the more safety significant results identified during recent PSRs.

**Answer** As far as the 2nd 10-yearly 1300 MWe safety review is concerned, the main results of the conformity studies or of the re-evaluation of the safety re-examination have resulted in the definition of 20 or so modifications, for instance:

- reinforcing the support for piping, cable trays and Component Cooling System tanks within the framework of the seismic re-evaluation of Safety and Electrical Buildings and the implementation of RFS 2001-01,
- an improvement in the electric power supply in to the injection pump at the primary pump seals in the event of the backed up electric supply panels failing,
- a change of the diesel engine cooling circuit regulating valves,
- the setting up of a logic control to trigger the primary pumps in the event of the loss of the cooling circuit and the seals.

Here are also some examples of backfitting measures as a consequence of the 2nd 10-yearly 900 MWe safety review (already in answer to question No 88):

- Examples of corrective actions of the first category are given below:
  - addition of insulation, of complementary heaters, protection of sensors, and definition of procedures to be applied in case of very low temperatures, due to a lower temperature for which safety related systems must be kept operating,
  - addition of sills and water level sensors in the sumps for internal flooding protection,
  - implementation of instrumentation to better monitor severe accidents such as containment pressure and hydrogen measurements
- In the second category, one can find:
  - Automatic make-up device to Reactor Coolant System (RCS) in case of loss of Residual Heat Removal (RHR) system.
  - Automatic interruption of dilution in case of primary pump trip and automatic switch-off suction from Chemical and Volume Control System (CVCS) pumps to Reactor Water Storage Tank (RWST) if the residual power is not sufficient for homogenisation.
  - Automatic isolation of CVCS let down line in case of a total loss of heat sink.
  - Redundant and diversified reactor trip signal to reduce the risk of high-pressure core melt.
  - Several improvements of operating procedures.
  - The programme of surveillance and maintenance of systems.

- Treatment of possible common mode failures on electrical power distribution.

| Seq. No 135 | Country | Article Article 14.2 | Ref. in National Report 2nd para.14.1.2 |
|---|---|---|---|

Question/ Comment

Reference: 14.1.2 (page 67) and 7.3.2.1.2 (page 33)
Q1: As France has 59 NPPs, it seems that there are approximately 1200 (59 x 20) inspections in a year. However, according to a table of 7.3.2.1.2 on page 33, number of inspections performed by the ASN ranges 330 to 380 per year. Would you explain these two different numbers of inspections?

Answer

In France there are 59 nuclear power reactors, which are located on only 19 different NPP sites. When an inspection goes to one NPP site it therefore concerns 1 to 6 reactors. Therefore about 20 inspections x 19 sites equal about 380 inspections per year.

Read also the answer to question No 97:
A minimal periodicity of inspection according to topics of a definite list is required for each BNI and each nuclear site, called “hard core”. According to this hardcore, each nuclear power plant is the subject of an average of about 15 inspections (depending of the number of reactor located on the nuclear site). An average of 3 to 6 additional inspections are performed following an event (depending of the event) and during outages (work site inspections).

| Seq. No 136 | Country | Article Article 14.2 | Ref. in National Report 14.1.3.1,Page 68 |
|---|---|---|---|

Question/ Comment

Reference :The purpose of the periodic safety reviews is therefore to reconsider the original safety demonstration and on the one hand to check that the installations are still in conformity with the initial requirements fixed for them and on the other to raise and improve their level of safety.

Q1: In the periodic safety review, are the software aspects of the safety regulatory requirements e.g. safety culture verified?

Answer

If "software aspects" means the operating procedures related to equipment, the answer is that their update according to equipment modification is verified.
If "software aspects" means the human behaviour and organisational factors, this is not checked during Periodic Safety Reviews but, either by routine inspection either by specific assessment performed by the Advisory Committee of experts for nuclear reactors.

| Seq. No 137 | Country | Article Article 14.2 | Ref. in National Report 14.1.3 |
|---|---|---|---|

Question/ Comment

We regard establishment of aging management measures are included in the scope of your performing of periodic safety review. We would like to ask the following questions intending to know about legal enforcement for the establishment of aging management measures and about penalty regulations if violations are confirmed. We would like to know about the degree of penalty quantitatively corresponding to the seriousness of the violation in such a way as how much franc the owner of nuclear power plant should pay or how many days the nuclear power plant should be suspended the commercial operation.

Q1: We would like to know the items, objects, outlines, regal enforcements and penalties about the periodic safety review in your country. The provision "Minister of Industry in association with the Minister in charge of prevention of serious technological hazard can require for the owners of the facilities to perform reexamination of safety of their facilities" in the article 5 of decree 63-1223 ( is this 1228 instead of 1223? ) of 11 December 1963 referred in 14.1.3.1 of your report provide the regal ground for the requirement of safety review, it does not contain the description of enforcement. What and how much penalties are fined if the periodic safety review is not performed against the provision?

Q2: In Japan, periodic safety review is performed in every 10 years after commercial operation started and technical assessment of safety for aged nuclear power plant (hereafter we call this assessment as "aging safety review") is required in addition to the periodic safety review if the plant operational year exceeds 30 years when the aging effects are regarded as to become serious. Is such aging safety review required in your country? We would like to know the items, objects, outlines, regal enforcements and penalties.

Q3: We would like to know about legal enforcement for the establishment of management program for maintenance of aged nuclear power plants including inspection, integrity assessment and repair/refurbishment (hereafter we call this program as "aging management program"). Also we would like to know about legal enforcement to confirm the execution of the program and results of execution. What penalties are fined when maintenance is executed not following the program, for example, the inspection frequency is less or the inspection extent is smaller than that defined in the program or evaluation method or repair technology is different from that defined in the program when defects are detected by the inspection in the components of the plant? If an accident is caused as a result of such maintenance as deviated from the aging management program, what penalties are fined? We would like to know examples of inside accusations which revealed the maintenance is executed against the aging management program though no accident is caused as a result of the faulty maintenance. What penalties are fined in such cases?

Q4: What penalties are fined when false descriptions are detected in the periodic safety review, the aging safety review and the report of the results of maintenance? What penalties are fined when a concealed defect by the false description caused an accident later? We would like to know examples of inside accusations which revealed the false description in the periodic safety review, the aging safety review and the report of the results of maintenance though no accident is caused. What penalties are fined in such cases?

Answer Q1 to Q4/

There is no legal enforcement in France for periodic safety review or maintenance planning. Legal enforcement exists if a license provides wrong information, or hide the right one, to the ASN inspectors.

Nevertheless, pressurized equipments' regulatory framework states that pressurized equipments must be submitted to severe and exhaustive testing every ten years (testing to 1,2 to 1,3 time the service pressure, exhaustive welding controls, other non-destructive controls). The authorization to use again the equipment is given after all the non-destructive controls are checked.

In case of non compliance, much more than a possible financial fine, which would remain very "symbolic" (some hundreds euros) as long as revision of the original law regulating nuclear activities is not completed, the effective penalty would be the lack of authorization to restart the reactor and the corresponding loss of earning for the operator, which could amount to a much higher order of magnitude than this possible fine.

| Seq. No 138 | Country | Article Article 14.2 | Ref. in National Report 14,1.3.1 |
|---|---|---|---|

Question/ Comment: Reference:14.1.3.1 General principles
The periodic safety review demands considerable resources on the part of the operator, but also on the part of the ASN and its technical support organization, the IRSN.

Q1: Regarding PSA, how long and how many man-hours does it take to complete for the IRSN assessment?

Answer: The time spent by IRSN in assessment of EDF's Probabilistic Safety Assessments (PSAs), not including IRSN's development time for its own research and assessment studies, which can be very important, is as follows:

- PSA 1 (version developed during studies conducted on the third 10-year outage safety reassessment for 900 MWe reactors): 2 engineers/year;
- PSA 2 (first Level-2 PSA developed by EDF in the same context): 3 engineers/year. This is the first time IRSN has assessed a Level-2 PSA.

Note: that IRSN's work on this assessment was facilitated by the fact that it had developed for years its own assessment studies.

| Seq. No 139 | Country | Article Article 14.2 | Ref. in National Report P78,79 |
|---|---|---|---|

Question/ Comment: Q1: Would you please describe review items regarding aging management in the 20-year and 30-year safety review on the 900 MW reactors?
Q2: Is there any different review item between the 20-year and 30-year safety review on the 900 MW reactors? If so, what is the reason?

Answer: Q1/
In terms of ageing, ASN has required EDF to present, for each plant, a file showing that it is able to be operated safely after 30 years lifetime. These files will be examined by ASN in the frame of the PSRs. They will include:

- the description of the installation and its operating conditions ;
- the information relating to the manufacture or the realization of the installation being able to contribute to the quantitative analysis of the mechanisms of ageing ;
- the assessment of the experience feedback of the behaviour of the installation in service ;
- the analysis of obsolescence risks ;
- the analysis of components or structures that are replaceable ;
- the list of ageing mechanisms.

Q2/
Yes, there are different review items between the 20-year and 30-year safety review on the 900 MW reactors. Indeed, the objective of the periodic safety review is also to focus on priority safety topics, which are dependent on operational experience feedback, new safety issues identified and the current development in science and technology. One of major differences is that for the definition of the scope of the 30-year safety review, ASN has asked EDF to make a comparison between EPR and the 900 MW plants in terms of safety.

Read also answer to question No 96:

Firstly, the 10 August 1984 order on quality (see report §7.3.1.3, p. 29) provides a general framework for provisions to be taken by any BNI operator to produce, obtain and maintain plant and operating quality standards compatible with safety requirements. This order is applicable to the studies performed in the frame of the periodic safety review.

As mentioned in the report (p. 68) the principles that regulate the conduct of periodic safety review are defined in the 11 December 1963 decree on BNIs (article 5). Then, on a case-by-case basis ASN issues letters to the BNI operators that define the scope of the safety review and specific issues that have to be considered.
For example, for 30-year safety review for 900 MWe, as mentioned in the report (§14.4.1.3, p. 74), the ASN issued a letter in October 2003 instigating the safety review, determining the scope and the limits of the studies to be made by EDF, together with the deadlines to be met to enable the resulting modifications to be integrated on the 900 MWe reactors during their third ten-yearly outages scheduled as of 2008. This letter has been made public on ASN's internet site: www.asn.gouv.fr. Hereafter are given the headlines of issues to be considered:

1/Internal and external hazards
- simultaneous failure of equipment non designed to withstand seismic conditions
- consideration of internal flooding in shutdown states
- internal explosion
- fire
- seismic verification approach
- Adverse weather conditions
- Hydrocarbon slick drift on river or seaside

2/Accident studies and radiological consequences
- cold overpressure
- Long term phases assumptions for accident studies
- Steam generator tube rupture
- Severe accident radiological consequences
- Containment
- Beyond design basis equipment
- Backup of Auxiliary Feedwater System tank
- Post accident surveillance information

3/Design of systems
- Design verification of civil engineering structures
- Functioning of Plant Radiation Monitoring system

- o Reliability of heat removal system of the fuel building
- o Performance of safety injection system
- o Reliability of emergency cooling recirculation function

| Seq. No 140 | Country | Article Article 14.2 | Ref. in National Report 14.2.2,1st.paragraph |
|---|---|---|---|

Question/ Comment This paragraph indicates that following the reviews performed every ten years, the applicable safety requirements are identified, guaranteeing their compliance by the units, and the safety analysis report is updated. As regards the participation of the regulatory authority in this approach, is this edition of the safety analysis report required to be individually approved for each installation? If this is not the case, is any type of supervision of these documents performed?

Answer The standard safety analysis report describes safety analyses generic to a series of reactors (900 MWe, 1300 MWe, 1450 MWe). The safety analysis reports of sites are supplementary reports to the standard one, which describe the specificities of sites and present primarily the analyses of safety relating to the external aggressions.
Both the standard and the site safety analysis reports are examined by ASN.

| Seq. No 141 | Country | Article Article 14.2 | Ref. in National Report Sect. 14.1.3.3 pg.69 |
|---|---|---|---|

Question/ Comment The report mentions that an order issued 10 November 1999 requires that, “after an operating period of 10 years, each main primary and secondary system of a pressurized water reactor undergo a full inspection and requalification comprising renewal of the hydrotest. The main primary system hydrotest, which consists in subjecting this system to a hydraulic pressure equal to 1.2 times the design pressure, constitutes an overall pressure resistance test...it enables identification of serious defects in unsuspected areas. Was this order retroactive in that all plants were required to conduct this test at their next refueling outages? Have any plants carried out this test in their 10-year outages since 1999, and if so, what were the results?

Answer Before the 10 November 1999 order, the 26 February 1974 order required this full inspection and requalification for each main primary circuit (but not for main secondary circuit). This 10 November 1999 order was not retroactive for main secondary circuit. All NPPs have already carried out this test (since 1974) on main primary circuits of but only some NPPs have already carried out this test (since 1999) on main secondary circuits. This test led to the detection of new degradation phenomena. For instance, in 1991, PWSCC on vessel head nozzles was detected during such an hydraulic test; this led later the operator to decide the vessel heads replacement on most of French PWRs.

**Art. 15 – Radiation protection**

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 142 | | Article 15 | page 77 |

Question/ Comment: Could you provide more detailed information on the criteria for determination of the NPP controlled and supervised area? Is additional division into sub-areas introduced within the controlled area?

Answer: EDF generally marks out the various zones of NPPs on the dose rate criterion only (green, yellow, orange and red) and regulates admittance to the orange and red zones. The provisions implemented to control contamination help reduce the risk of internal exposure to well below 1/100 of the Derived Air Concentration (DAC) and therefore render it negligible. This risk is taken into consideration and processed specifically by site.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 143 | | Article 15 | page 78 |

Question/ Comment: Could you provide more details about the methods for population equivalent and effective dose calculation and about the methods to be used for assessment of the impact on the population, determined in the ministerial order dated 01.09.2003?

Answer: The general principles for population equivalent and effective dose calculation and about the methods to be used for assessment of the in of the impact on the population, which are determined in the ministerial order dated 01.09.2003, are derived from the European directive 96/29 (available on European Commission website: http://europa.eu.int/eur-lex).

They states that effective doses shall take into account the various exposure pathways from an atmospheric cloud, from drowning into water and from sol deposits. This exposure dose is the product of the activity of the radionuclides present in the ambient atmosphere by the external dose coefficient for the same radionuclides.

These methods are used to determine the impact of nuclear installations' releases on reference groups within the population from modelling transfer path of rafionuclides to human body.

The full text of the methods together with the tables for various radionuclides coefficients represent around 60 pages which cannot be reproduced here.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 144 | | Article 15 | page 87 |

Question/ Comment: What are the maximum individual doses registered at NPP in 2003?

Answer: EDF uses an internal computer system common to all the sites equipped with nuclear reactors to record the dose rate accumulated over the same year by people working on these installations. This system offers effective prevention. In 2003, 1 person reached 22 mSv over 12 months, 3 exceeded 18 mSv over 12 months at the end of 2003, 67 exceeded 16 mSv over 12 months. This concerned EDF personnel or contractors. Generally speaking, these operators reach these levels working successively on the various EDF sites.

Look also at results in 2004, given in answer to question No 156:

The average annual individual dose rate for an operator working on EDF reactors (EDF employee or one of its contractor employees) was 1.7 mSv in 2004. At the end of 2004, none of the operators exceeded 18 mSv over 12 months. 34 people exceeded 16 mSv over 12 months.
27000 people were involved and received a non-zero dose on EDF reactors in 2004.

| Seq. No 145 | Country | Article<br>Article 15 | Ref. in National Report |
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| Question/ Comment | Which acceptance criteria have been used for the regulatory review of the radiological consequences of design basis accidents? Are these criteria related to releases or related to radiological exposures? If dose limits are applied, which are the parameters (e.g. exposure pathways, integration times, distances) considered for the calculation? |
| Answer | Review of the design basis accidents: the radiological "intervention levels" have been established by the Ministry of health in 2000 (and are defined in a ministerial order since 2003). The main parameters are the following:<br>- Pathways: exposure, contamination, inhalation;<br>- Calculation integration time: 24h ;<br>The characteristic distances are defined from the comparison "intervention levels" / "consequences of basis accidents". |

| Seq. No 146 | Country | Article<br>Article 15 | Ref. in National Report<br>Para 15.1.2.2 - p.77 |
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| Question/ Comment | It is noted that the notion of clearance threshold has not been adopted in France. How therefore does France intend to address the issue of clearance of waste materials, e.g. steel and concrete, arising from the decommissioning of nuclear power plants?. |
| Answer | The clean-up method favoured by the ASN for nuclear installations is based on a waste zoning methodology founded on successive and independent lines of defence (see report p. 123). As the first line of defence, using a demonstration based on the design of the installation, its operating methods, an analysis of its history (incidents, modifications, periodic radiological checks, etc) or any other empirical type of demonstration, the operator must determine the zoning of the waste in its installation by accurately defining the boundary between conventional waste zones and nuclear waste zones. In the particular case of building walls, this boundary can correspond to a minimum clean-up thickness. The operator then removes all the nuclear waste from the nuclear waste zones, before implementing, as a second line of defence an appropriate inspection program on the remaining items, to confirm that they are indeed non-radioactive.<br>The operator then proposes to the ASN to consider the remaining zone as a conventional waste zone. After approval of this final waste zoning modification by the ASN, the remaining conventional waste is disposed of in conventional routes and can be dealt with in the same way as normal industrial waste. |

The French approach about clearance, which relies on a case by case basis assessment, has been addressed in the 1st France Report for the Joint Convention (May 2003) and will be repeated in its 2nd report (October 2005)

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 147 | | Article 15 | 15.2.1,p81, 82 |

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| Question/ Comment | Reference: EDF has succeeded by an ALARA policy to reduce worker's collective dose from 2.4 man.Sv/year per unit in 1992 to 1.08 in 2000, and to 0.89 in 2003.<br>Q1: Among the countermeasures taken for reducing the dose, what was the most effective countermeasure?<br>Q2: From the experience gained since 1992 in this field, which avenue among the three will contribute most in reducing the collective dose?<br>Q3: The new ALARA approach might have cost a lot. How does EDF appraise the cost benefit of the new ALARA approach? |
| Answer | Q1/<br>The most important measure is the implication of the management to support the Alara approach and create a real Alara culture among the personnel and contractors. When the dynamism gets this kind of impluse at the highest level, ideas in the field proliferate. In concrete terms at EDF, the decision to create an Alara committee on each nuclear site, headed by a director and to create a national committee, played an important role in the first phase of Alara.<br><br>Q2/<br>During the first 10 years, EDF focused as a priority on 10 or so outage sites representing 80% of the dose rate and made considerable progress. We can gain even more on these large sites (example of testing and Steam Generators works) but gains in terms of the dosimetry are less spectacular. Under these conditions, we have to process all the other activities under development for a systematic approach to analysis even if it is graduated. This is the purpose of approach No. 2. To continue its progress, EDF is developing its dosimetric information system to provide activity-preparing officials with a computer tool to prepare for activities in terms of radioprotection. This tool will be common to the contractors.<br><br>Q3/<br>The optimization approach is a regulatory obligation. It is one of the major expectations of the EDF staff and the contracting company staff. The choice of the major outlines has not been the subject of overall cost-profit analysis. The choice and evaluation of the themes providing the greatest progress are based on analysis and on the opinion of managers and internal experts. EDF is also taking into consideration international experience feedback. In this way, reduction of the source term appears to be one of the main lines of development among the many foreign PWR operators (zinc injection, shutdown optimization, choice of filters and resins...), and EDF has entered it into its action plans. the cost and profit approach, and and multi criterion analysis, and are used for decisions regarding targeted actions. For instance, modifications to installations with a radio-protection aspect that can be carried out during the third 10-year inspections of PWR 900 MW reactors have been classified in order of interest. The tools used for these analyses have a decision-making aid status, up to the manager on the basis of all the evaluation elements available to him. |

| Seq. No<br>148 | Country | Article<br>Article 15 | Ref. in National Report |
|---|---|---|---|

Question/ Comment

In relation to 15.5.1, it is stated that for contractor staff of EDF and CEA, the monitoring is conducted by IRSN and LCIE.

1. Who is responsible for dose management for contractor staff?

2. According to dose record at reactor facilities for the year 2002, five contractor workers received annual dose that exceeded 20 mSv. What are major reasons for the overexposures and what are measures for the reduction of the doses of these contractor workers?

Answer

In accordance with the regulations, the employer is responsible for the management of the dose to which the staff is exposed. The employer gets in touch with IRSN or an approved organization like LCIE to measure the external passive exposure (at present by film). As far as the operational dosimetry readings are concerned (electronic dosimeter), EDF supplies an electronic dosimeter to each participant (on its staff or a contractor's) but this method (a possibility offered by regulations) does not relieve each employer of its responsibility with respect to its staff.

As concerns the exceeding of a 12-month total dose rate of 20 mSv as registered in 2002, two years later we cannot re-establish the causes. We should bear in mind that at that date, the regulatory limit was 50 mSv.

To avoid total exposures of more than 20 mSv over 12 months: EDF uses an internal computer system common to all the sites equipped with nuclear reactors to record the dose rate accumulated over the same year by people working on these installations, making it possible to monitor the development of 12 month doses received by every participant on an EDF site. Before authorising an operator to enter a controlled zone, a check is run on his/her twelve month total dose. If this rate exceeds the alarm threshold set at 16 mSv, an alert is given and the direct manager of the person is consulted. If a second alarm threshold set at 18 mSv is reached, the person is temporarily prohibited from entering the controlled area until a specific monitoring file has been drawn up. This system offers effective prevention. In 2003, 1 person reached 22 mSv over 12 months, 3 exceeded 18 mSv over 12 months at the end of 2003, 67 exceeded 16 mSv over 12 months.

For its part, the CEA recalls that the employer, in particular when he is not the establishment head wherein the workers are exposed to ionising radiation, is responsible of the management of doses received by these workers. But, in application of the optimisation principle, the head of establishment manage the operational dosimetry data linked to operations of all workers (external contractors included) performed in the controlled areas of the establishment. This operational dosimetry, for contractor's agents, acts as complement to the dosimetry under responsibility of the employer. The French Labour code describes the responsibilities and the circulation of the information about dosimetry data, between the establishment head, the employer and the person with radiation protection competence. In 2002, for CEA, a maximum dose of 8.2 mSv was received by a contractor agent working in the cleaning field.

| Seq. No 149 | Country | Article Article 15 | Ref. in National Report page 77-78 |
|---|---|---|---|

Question/ Comment: Are there any plans to change French approach to clearance because this option is foreseen in EC legal documents? (section 15.1.2.2)

Answer: The clearance threshold provided for in EC legal document is only an option.

France does not see any reason to change its current approach based on a case by case assessment which has been addressed in the 1st France Report for the Joint Convention (May 2003) and will be repeated in its 2nd report (October 2005)

| Seq. No 150 | Country | Article Article 15 | Ref. in National Report Section 15.1.4 |
|---|---|---|---|

Question/ Comment: Section 15.1.4 of the National Report (as well as section 7.2.1) stresses the need for obtaining separate permits for water intake as well as for liquid and gaseous effluents.
1) What are the reasons for the need to have separate licenses/permits both for operation and water intake as well as for liquid and gaseous discharges?
2) Is it possible to operate a plant if you only have an operation license and do not have permits for water intake and liquid-gaseous discharges?

Answer: Q1/
The report does not mention any "separate permits" but to the contrary stresses on the two aspects (intake and release) of the same single permit.

Q2/
In France, legal basis order one single permit both for water intake as well as for liquid and gaseous discharges. Without such a permit, operator can't operate.

| Seq. No 151 | Country | Article Article 15 | Ref. in National Report Section 15.2.1 |
|---|---|---|---|

Question/ Comment: Good Practice:
EdF have alarm thresholds when doses get to 16-18 mSv. Individuals who are approaching the 20 mSv limit can thus be more closely monitored to prevent them from exceeding this limit.

Answer: France is thankful for this comment

Read also answer to question No 148:

In accordance with the regulations, the employer is responsible for the management of the dose to which the staff is exposed. The employer gets in touch with IRSN or an approved organization like LCIE to measure the external passive exposure (at present by film). As far as the operational dosimetry readings are concerned (electronic dosimeter), EDF supplies an electronic dosimeter to each participant (on its staff or a contractor's) but this method (a possibility offered by regulations) does not relieve each employer of its responsibility with respect to its staff.

As concerns the exceeding of a 12 month total dose rate of 20 mSv as registered in 2002, two years later we cannot re-establish the causes. We should bear in mind

that at that date, the regulatory limit was 50 mSv.
To avoid totals of more than 20 mSv over 12 months: EDF uses an internal computer system common to all the sites equipped with nuclear reactors to record the dose rate accumulated over the same year by people working on these installations, making it possible to monitor the development of 12 month doses received by every participant on an EDF site. Before authorising an operator to enter a controlled zone, a check is run on his/her twelve months total dose. If this rate exceeds the alarm threshold set at 16 mSv, an alert is given and the direct manager of the person is consulted. If a second alarm threshold set at 18 mSv is reached, the person is temporarily prohibited from entering the controlled area until a specific monitoring file has been drawn up. This system offers effective prevention. In 2003, 1 person reached 22 mSv over 12 months, 3 exceeded 18 mSv over 12 months at the end of 2003, 67 exceeded 16 mSv over 12 months.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 152 | | Article 15 | Section 15.4.2 |

Question/ Comment: Good Practice:
EdF Nuclear Power Plants have a separate department on site keeping regulatory registers (effluents and environmental) independently of the department carrying out releases and is directly answerable to the plant manager.

Answer: France is thankful for this comment

Yes, every NPP has this organization

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 153 | | Article 15 | 15.2.1,par. 3,pag.80 |

Question/ Comment: Point 15.2, Measures taken by EDF in the area of Radiological Protection, indicates that in order to achieve the target dose of 0.8Sv-person/year in 2005, a new ALARA approach with three fundamental aspects has been launched, but no mention is made of training in RP.
How are the radiological protection training requirements for the plant professionally exposed personnel and for off-site or contracted workers defined and established?

Answer: Two training levels are required for work in positions exposed to ionising radiation at EDF. A basic level referred to as RP1 considered essential for the operator to circulate alone in a controlled area or to work under the responsibility of a supervisor and a level RP2 for the supervisor in charge of a working team. RP1 training is a five-day course. RP2 training is an additional five-day course.

The specifications for this training for the contracting companies are established by a professional approved organisation known as CEFRI. This organisation issues a certificate to the training companies.

For its personnel, EDF has its own in-house training service but the nature and content of the training are the same.

In accordance with the regulations, this training is followed up by a refresher course every three years.

| Seq. No 154 | Country | Article Article 15 | Ref. in National Report page 77-78 |
|---|---|---|---|

Question/ Comment: Some special dose limits are missing in the report, e.g. dose limit for young workers (below 18 y)
It is not described, how the dose limit of 1 mSv for the public is not exceeded, if there are several facilities as sources for exposure.
Please give the missing information.

Answer: Q1/
Limits for young workers were not given in the report since, in practice, these workers are not allowed to work in the nuclear installations under the scope of this Convention.

Effective dose for young workers (16-18 years old) is stated at 6 mSv/year, but in practice they are not allowed to work in controlled area.

Q2/
Since there may be several facilities at sources for exposures, the assessment of each one require that the impact to the public be lower as reasonable as possible (ALARA), therefore there is margins to be sure to comply with the 1 mSv criteria with several sources of exposures. However, France does not use the concept of dose constraint.

| Seq. No 155 | Country | Article Article 15 | Ref. in National Report page 79 |
|---|---|---|---|

Question/ Comment: Please describe the conditions for clearance of inactive or low level radioactive material from nuclear sites.

Answer: The French approach about clearance, which relies on a case by case basis assessment, has been addressed at length in the 1st France Report for the Joint Convention (May 2003) and will be repeated in its 2nd report (October 2005).

In summary, in France, a specific system, not implementing any clearance level, has been put in enforcement in order to ensure a safe and transparent waste elimination system for waste generated by nuclear facilities. This system is founded on successive lines of defence based on the elaboration of a "waste zoning" corresponding to a segregation between "nuclear waste" (waste susceptible to be or to have been contaminated by radionuclides or activated) and "conventional waste" (waste that is not susceptible to be or to have been contaminated nor activated – see report p.123). This management of radioactive waste from basic nuclear installations is structured within a strict regulatory framework, defined by a ministerial order of 31 December 1999 stipulating the general technical regulations intended to prevent and limit the detrimental effects and external hazards resulting from the operation of basic nuclear installations.

"Nuclear waste" has to be eliminated in dedicated facilities or repositories, or in conventional facilities under the condition of a special authorisation based on a radiological impact study and a public inquiry. "Nuclear waste" is, as a safeguard, considered to be at least "very low level" waste. As a consequence, a disposal site for this type of waste was necessary. A VLL waste repository was authorised (as an installation classified on environmental protection grounds) by the Aube department prefect in 2003."Conventional waste" is disposed of in conventional routes. Basic traceability is in all cases guaranteed.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 156 | | Article 15 | 15.5.1,P.87 |

Question/ Comment: In the table, it was stated that the individual doses are below 20 mSv. If it is available, would you provide maximum and average individual dose values for NPPs?

Answer: The average annual individual dose rate for an operator working on EDF reactors (EDF employee or one of its contractor employees) was 1.7 mSv in 2004. At the end of 2004, none of the operators exceeded 18 mSv over 12 months. 34 people exceeded 16 mSv over 12 months.
27000 people were involved and received a non-zero dose on EDF reactors in 2004.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 157 | | Article 15 | 15.2.2.1 p.82 |

Question/ Comment: The National Report states that EDF claims the "dose-related impact of radioactive releases remains extremely low at less than 1 µSv/year". Is this the dose to the most exposed member of the public, or an average for the French population? Would the doses attributable to non-reactor sites such as La Hague be significantly different?

Answer: The annual radiological release impact was the subject of a regulatory request in the new release application permits for NPPs. The method of calculation used in establishing this impact was radioelement by radioelement, taking into consideration the specific data of each site (population groups concerned, eating habits, weather conditions, etc). The impact is measured in micro-Sievert/year.

As regards La Hague plant, though it is not within the scope of this Convention, France is pleased to provide its evaluation of the maximal impact of annual releases expressed in terms of effective dose on "reference groups" i.e. on group of members of the population on which exposure coming from a given source is relatively homogeneous and which are representative of the people receiving the highest doses coming from these sources:

Limits (Order 1984): 0.120 mSv – Actual release 1999: 0.011 mSv
Limits (Order 2003): 0.020 mSv – Actual release 2004: 0.010 mSv

This means that the impact is very slightly higher than that of a NPP.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 158 | | Article 15 | 15.5.1 p.82 |

Question/ Comment: In order to provide a better indication of how doses are falling over time at French nuclear sites, would France consider expanding the table on this page to show the numbers of workers exceeding, say 5mSv, 10mSv and 15mSv?

Answer: The most recent datas published are related to the year 2003, refer to slightly different ranges and concerns not only workers from NPP but all nuclear facilities: within 69512 people monitored, the exposures were as follows:

- From 0 to 1 mSv (limit for the public): 60 153 (86,5%)
- From 1 to 6 mSv : 7 568 (10,9 %)

- From 6 to 20 mSv (new limit for workers): 1 778 (2,6 %)
- From 20 à 50 mSv : 13 (0,02 %)
- Above 50 mSv : 0

**Art. 16 – Emergency preparedness**

| Seq. No 159 | Country | Article Article 16 | Ref. In National Report |
|---|---|---|---|

Question/ Comment: What were the findings and recommendations concerning the emergency organisation of the OSART missions at Nogent and Civaux NPPs?

Answer: <u>OSART Civaux:</u>

The Emergency Planning and Preparedness topic gave rise to 2 Good Practices and 1 suggestion, related to the following issue: The process of gathering and counting people is not really efficient.

- Staff counting on site is by hand, generating a delay of between 1 and 1.5 hours before the first overall evaluation of the situation is available.
- The steps needed to activate the seven gathering points in the buildings mean moving equipment (radioprotection markers, etc.) from the crisis rooms on site near the main entrance to the gathering points concerned.
- There are no general or dedicated resources for the regular releasing of information at the gathering points. Useful information regarding the change in the situation is provided by the PCM (Resource Control Post) by a telephone call to the person in charge of the gathering point. Subsequently, this Post supplies the information received by megaphone.
- In spite of the good marking and the indications given at the seven gathering points to be used in cases of radiological emergency, and the use of signs and identifications that are comparable with those of the other gathering point (those to be used in case of fire or medical emergency), - six points set out over the entire site -, there is a risk of triggering confusion between the two types of gathering points. In addition, in at least one case, the outdoor gathering point is too close to the fire hydrants to be used by the firemen. This is liable to interfere with the intervention teams.

Note that the staff at Civaux has already undertaken a number of actions at the time of the mission to resolve some of the cases listed above by using at the outset the devices in place at other EDF plants (Blayais). If personnel gathering and counting in an emergency are not efficient, it is liable to cause their pointless dose exposure and/or a response that is unsuited to the crisis situation.

NOTE: An observation similar to that of point 1 was made during the recent OSART on the Penly site – see the EDF position in the following answer to question No 167 about processing:

Access controls to an area in an emergency situation, internal to the site, come under national prescriptions specific to an operator asking at the gate for access to a nuclear site, to minimize in and out movements to authorized people. In particular, in and out movements must be confined to on-call staff and external emergency people. All other access must be approved by the emergency manager.

As far as counting personnel on the site in an emergency is concerned, the subject of suggestions during OSART missions, EDF is looking at how to use access control computer data automatically. However, regulatory requirements have to be resolved to develop this kind of data use: Indeed, these are part of the site protection aspect (safety) and are concerned by the "computers and liberty" regulations. EDF also intends to get in touch with operators identified by IAEA to refer to their good practice rules.

OSART Nogent:
The Emergency Planning and Preparedness topic gave rise to 1 Good Practice and 1 suggestion, related to the following Issue:
Some emergency exercises and training activities miss opportunities to enhance the knowledge of participants and improve emergency response. Some examples of these are:

- Results of nuclear emergency exercise showed a problem for communication and evacuation of on-site people.
- Regular emergency exercises (4-6 times/year) are conducted using the control room. It would be better to keep the control room calm and to develop other areas where operators can be more involved. (The simulator at Cattenom is only used every 18 months for on-site and off-site exercises).
- Concerning the exercise for evacuation of onsite people, the accounting of evacuated people will take 1-1.5 hours, which is longer than good international practices.
- There is no (white) board or other highly visible communications aid in the LTC (Local Operations Emergency Centre). Boards are necessary to have a discussion or share information among emergency response people.
- Staff responsible for ELC (Local Emergency Response Team) was not completely familiar with the content of analysis done in EDF national as he was recently appointed to this position.

If all aspects of the emergency planning and practices arrangements are not comprehensively conducted, opportunities to improve individuals' competence (preparedness for an emergency situation) may not be assured.

Since these OSART missions, OSART follow-up missions have occurred (end 2004) and considered that these issues were now solved at both NPPs.

The full OSART reports are publicly available on the ASN website
These 2 OSART Follow-up reports should also be available by May 2005.

| Seq. No 160 | Country | Article Article 16 | Ref. in National Report |
|---|---|---|---|

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| Question/ Comment | Which body in France ensures the function of National Warning Point under the international conventions ? |
| Answer | The Ministry of Foreign Affairs ensures the function of National Warning Point under the international conventions. DGSNR ensures the function of Competent national authority. |

| Seq. No 161 | Country | Article Article 16 | Ref. in National Report |
|---|---|---|---|

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|---|---|
| Question/ Comment | What does it mean the „vicinity of the site concerned“? Are that for instance circles around the NPP? If yes – what their radius and how, on which basis, were they established ? On whose expenditures are the iodine tablets ensured ? |
| Answer | The off-site emergency plan (PPI) of a BNI is defined on the basis of the |

consequences of the accidents described in its on-site emergency plan.
Each PPI describes the countermeasures to be taken to protect population and goods around the site (NPP or others).

The consequences of the accidents are compared with “decision levels” (radiological intervention levels, IDLH…). This comparison leads to the definition of circles around the sites. The radius depend on the characteristics of the site. For NPP, 3 radius are considered :

- Sheltering : 10 km ;
- Evacuation : 5 km ;
- Iodine preventive distribution : 10 km.

Iodine matters : the financing of iodine tablets is ensured by the operator within the given PPI circle. Beyond this area, the Ministry of health ensures the financing.

Read also Answer to questions No 162 :

The new preventive distribution of iodine tablets campaign due in 2005 involves a personal letter addressed to each family living around NPP within a radius of 10km. This letter includes an individual numbered withdrawal paper to seek tablets at nearest pharmacies. A complementary distribution will be organised in order to provide tablets to the family which haven’t seek tablets to pharmacy.

Read also Answer to questions No 163 for more information:

The latest iodine tablets preventive distribution campaign enabled 2 main means of distribution :

- Door-to-door distribution ;
- Mailing of a withdrawal coupon to swap for iodine tablets in a pharmacy.

The most efficient mean is the door-to-door distribution.
Moreover, France plans the stockpiling of iodine in each department with a view to improving provisions for the protection of children, adolescents and young adults against radioactive iodine beyond the PPI zone.

| Seq. No 162 | Country | Article Article 16 | Ref. in National Report Paragraph 16.5.2 |
|---|---|---|---|

Question/ Comment: It is noted that following the experience gained from accident drills between 1995 and 1996, it was decided to opt for the preventative distribution of iodine tablets to populations living near nuclear power plants. Does this involve actually distributing iodine tablets to individual families or to appropriate outlets, e.g. pharmacies, or to both. Also, in the case of the general population, for which age groups are stable iodine tablets recommended?

Answer: The new preventive distribution of iodine tablets campaign due in 2005 involves a personal letter addressed to each family living around NPP within a radius of 10km. This letter includes an individual numbered withdrawal paper to seek tablets at nearest pharmacies. A complementary distribution will be organised in order to provide tablets to the family that have not seek tablets to pharmacy.

| Seq. No 163 | Country | Article Article 16 | Ref. in National Report Paragraph 16.5.2 |
|---|---|---|---|

Question/ Comment It is noted that after the completion of preventative distribution the drill sessions revealed the need for "further improvements in this respect". What improvements have been indicated as necessary to ensure that all members of the public, who should take stable iodine tablets following a release of radioiodine, do in fact take them?

Answer The latest iodine tablets preventive distribution campaign enabled 2 main means of distribution:

- Door-to-door distribution;
- Mailing of a withdrawal coupon to swap for iodine tablets in a pharmacy.

The most efficient mean is the door-to-door distribution.
Moreover, France plans the stockpiling of iodine in each department with a view to improving provisions for the protection of children, adolescents and young adults against radioactive iodine beyond the PPI zone.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 164 | | Article 16 | |

Question/ Comment Please explain who and how determines the emergency planning zone around the nuclear facilities. What is the basis for the zone specification (engineering judgement)? Which criteria do you apply to specify the zone area? Do you use/accept any probabilistic arguments to determine the zone area?

Answer The technical grounds of the off-site emergency plan (PPI) are defined by the DGSNR, on the basis of the information given by the operators. The circles are demonstrated according to the assessment of the consequences of several accident's scenarios.
Probabilistic arguments are not used to choose the convenient scenarios.

Read also the answer to question No 161:

The off-site emergency plan (PPI) of a BNI is defined on the basis of the consequences of the accidents described in its on-site emergency plan.
Each PPI describes the countermeasures to be taken to protect population and goods around the site (NPP or others).
The consequences of the accidents are compared with "decision levels" (radiological intervention levels, IDLH…). This comparison leads to the definition of circles around the sites. The radius depend on the characteristics of the site. For NPP, 3 radius are considered :

- Sheltering : 10 km ;
- Evacuation : 5 km ;
- Iodine preventive distribution : 10 km.

Iodine matters : the financing of iodine tablets is ensured by the operator within the given PPI circle. Beyond this area, the Ministry of health ensures the financing.

Then, the prefects of the departments concerned determine the definitive planning zone, taking into account local general issues (major wishes, for instance).

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 165 | | Article 16 | section 16.3.1, p 97 |

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| Question/ Comment | Last paragraph in the section states that the technical emergency response teams not only assess the situation, but also predict how it will develop.<br>How the uncertainties in the predictions are taken into account while recommending protective actions and how the predictions are corrected if the measurements do not match the predictions? |
| Answer | The assessment of the progression of the accident, notably the calculation of radiological consequences, enables the prefect to bring the population's protection actions forward.<br>The uncertainties are explained to the prefect.<br>Then, according to the information given by the operator or measurements in environment, corrections can be brought:<br>- if the predictions were over-assessed, the preparedness of the protection actions can be stopped ;<br>- in case the predictions should have been under-assessed, protective actions are widened. |

| Seq. No 166 | Country | Article Article 16 | Ref. in National Report pages 88-104 |
|---|---|---|---|

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| Question/ Comment | What specific radiological protection training is received by the people acting locally in the off-site emergency plans (firemen, medical personnel, police) and who is responsible for delivering this training? What radii are defined by the PUIs and PPIs? |
| Answer | People acting locally in the off-site plan (firemen and medical body specialised in radiation protection) are given specific radiation protection during their education by a wide range of competent bodies including the DGSNR.<br><br>For EPZ radii, read answer to question N° 161:<br>The off-site emergency plan (PPI) of a BNI is defined on the basis of the consequences of the accidents described in its on-site emergency plan.<br>Each PPI describes the countermeasures to be taken to protect population and goods around the site (NPP or others).<br>The consequences of the accidents are compared with "decision levels" (radiological intervention levels, IDLH…). This comparison leads to the definition of circles around the sites. The radius depend on the characteristics of the site. For NPP, 3 radius are considered :<br>- Sheltering : 10 km ;<br>- Evacuation : 5 km ;<br>- Iodine preventive distribution : 10 km. |

| Seq. No 167 | Country | Article Article 16 | Ref. in National Report |
|---|---|---|---|

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|---|---|
| Question/ Comment | Why are access controls not established among the protective measures to be implemented in the event of emergency situations? |
| Answe | Access controls to an area in an emergency situation, internal to the site, come under national prescriptions specific to an operator asking at the gate for access to |

a nuclear site, to minimize in and out movements to authorized people. In particular, in and out movements must be confined to on-call staff and external emergency people. All other access must be approved by the emergency manager. As far as counting personnel on the site in an emergency is concerned, the subject of suggestions during OSART missions, EDF is looking at how to use access control computer data automatically. However, regulatory requirements have to be resolved to develop this kind of data use: Indeed, these are part of the site protection aspect (safety) and are concerned by the "computers and liberty" regulations. EDF also intends to get in touch with operators identified by IAEA to refer to their good practice rules.

Read also the answer to question No 159 related to the 2003 Civaux OSART emergency planning and preparedness findings:
The process of gathering and counting people is not really efficient.

- Staff counting on site is by hand, generating a delay of between 1 and 1.5 hours before the first overall evaluation of the situation is available.
- The steps needed to activate the seven gathering points in the buildings mean moving equipment (radioprotection markers, etc.) from the crisis rooms on site near the main entrance to the gathering points concerned.
- There are no general or dedicated resources for the regular releasing of information at the gathering points. Useful information regarding the change in the situation is provided by the PCM (Resource Control Post) by a telephone call to the person in charge of the gathering point. Subsequently, this Post supplies the information received by megaphone.
- In spite of the good marking and the indications given at the seven gathering points to be used in cases of radiological emergency, and the use of signs and identifications that are comparable with those of the other gathering point (those to be used in case of fire or medical emergency), - six points set out over the entire site -, there is a risk of triggering confusion between the two types of gathering points. In addition, in at least one case, the outdoor gathering point is too close to the fire hydrants to be used by the firemen. This is liable to interfere with the intervention teams.

Note that the staff at Civaux has already undertaken at the time of the mission a number of actions to resolve some of the cases listed above by using at the outset the devices in place at other EDF plants (Blayais). If personnel gathering and counting in an emergency are not efficient, it is liable to cause their pointless dose exposure and/or a response that is unsuited to the crisis situation.

After the Civaux OSART follow-up mission, end 2004, the issue was considerd as resolved.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 168 | | Article 16.1 | §2.3.6-P.10,P16,P.96 |

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| Question/ Comment | Does the Nuclear Safety Authority plan to regulate the preparedness for extreme meteorological situations (e.g. extreme cold or hot)? |
| Answer | Yes, following the 1999 Blayais NPP and 2003 Cruas NPP emergency situations, the work is in progress. |

| Seq. No 169 | Country | Article Article 16.1 | Ref. in National Report Art. 16, P. 88 |
|---|---|---|---|

Question/ Comment: How the emergency preparedness is managed during the decommissioning phase of nuclear facilities?

Answer: With respect to the management of emergency situations, the organization at EDF for dismantling is based for the greater part on the organization already in place for the installations in operation: There is only one on-site Emergency Plan (PUI) per site, covering all the installations whether they are in the process of dismantling or in operation (case of Bugey, Saint Laurent, Chinon and Chooz). For the two isolated sites (Creys and Brennelis) the organization is as per the general organizational outline as the other sites, adapted to the risk and drawing whenever necessary on the existing national level, in particular for the support of the national Emergency Preparedness team. The teams dedicated to dismantling on the sites are integrated into this organization and complete it.

| Seq. No 170 | Country | Article Article 16.1 | Ref. in National Report 16.4, 16.5.3, |
|---|---|---|---|

Question/ Comment: Reference : 16.4(page 100-101), 16.5.3(page 100)
France is developing emergency preparedness very well. Roles and responsibilities of each relevant staff and organization are clearly defined.
In the field of emergency drills, which is very important in order to test the described emergency response provisions, it should be noted that national emergency response drills involving the relevant local residents are organized yearly, that international drill sessions and co-operation are reinforcing together with neighbouring countries, that post-accident management drills are also taking into account, and especially, that lessons learnt are well analyzed and feed-backed.
Q1: Please explain about the feed-back system of lessons learnt from emergency drills.
Q2: What does the wordings “environmental clean-up” mean?

Answer: Q1/
The feed back consists in meetings of the people involved in the drills according to the following schedule:
- D-Day: Exercise + immediate feed-back;
- D+15: local feed-back;
- D+30: general feed-back;
- Year+1: final assessment (if needed)

Moreover, each third month, a general drills feedback meeting is organised by the DGSNR. If a lesson learned from a drill requires any evolution in emergency organisation, this is stressed and closely followed during these kinds of meetings.

Q2/
Environmental clean-up means environment’s decontamination management

| Seq. No 171 | Country | Article Article 16.2 | Ref. in National Report 16.4.2,P.101 |
|---|---|---|---|

Question/ Comment: More information would be appreciated on scope and extent of exchanged information between the France and its neighbouring countries for the emergency planning purposes.

Answer France has signed bilateral agreements for exchanging information in case of emergency with its neighbouring countries close to one of its basic nuclear installation: Luxembourg, the United Kingdom of Great Britain and Northern Ireland, Belgium, Germany, Switzerland. These agreements establish the existence of a contact point available 24h in each country and the possibility of exchanging information by the local authorities. In application of these agreements, specific protocols described the type of information to exchange. Annual or bi annual meetings between the competent authorities for applying these bilateral agreements allow extensive exchange about participation of the neighbouring countries in national exercises, emergency planning and practices, public information.

**Art. 17 - Siting**

| Seq. No 172 | Country | Article Article 17 | Ref. in National Report |
|---|---|---|---|
| Question/ Comment | Flamanville has been selected the site for siting an EPR. What are the socio-economic aspects decisive to choose this site? Which safety aspects have led to also give preference to this site? | | |

Answer

The Flamanville site was chosen by EDF according to 4 criteria:

- the availability of the real estate reserves;
- the absence of any specific environmental demands;
- the capacity to evacuate the electricity produced on the commissioning date;
- the site accommodation conditions;

These 4 criteria result from technical feasibility on the one hand and from the concern of EDF to establish a commissioning date from the new unit as early as possible.

Availability of real estate reserves.
To save on time related to the acquisition of complementary land (negotiation, declaration of public utility, expropriation, etc), EDF has preferred the site at Flamanville where there is already land for the construction of two new units. Indeed, during the construction of the first two units commissioned in 1985 and 1986, the site was prepared with a view to accommodating four production units.

The absence of any specific environmental demands.
Because of its situation of on the coast of the channel, Flamanville has a considerable cooling capability, not requiring the construction of a cooling tower. Impact studies on the environment date back to 1995 and do not reveal any specific problems. They will be updated by a new sampling campaign.

The capability of evacuating the electricity generated.
The commissioning of the new electricity production unit presupposes that the Very High Voltage (VHV) transmission capabilities of the electricity produced will indeed be available at the desired date. None of the sites having a real estate reservations offers this capability in reality. Therefore, the company has examined this aspect with particular attention and is asking RTE (the organisation managing of the high-voltage electricity transmission network postbag its for its expertise, estimating that all of the power produced can be evacuated if the VHV network around Flamanville is strengthened. RTE will be processing this project in order to bring it into compatibility with the commissioning of Flamanville 3.

The site accommodation conditions.
The setting up of the new production unit would be enhanced by the backing of the local players and if it is properly understood by the entire population: the quality of reception extended to the project is therefore one of the conditions for success, both for public debate and for the site which will be a large scale one, and of course, for the operation of the new plant. The Flamanville 3 project has been the subject of strong consensus among the local elected authorities and the local economic players.

| Seq. No 173 | Country | Article Article 17 | Ref. in National Report |
|---|---|---|---|

Question/ Comment: One of the compulsory constraints on selecting a site is the consideration of the site's environment. What are the considerations concerning the site's environment that were evaluated and what were the most prominent results?

Answer: From the outset, the Flamanville site was planned for the accommodation of 4 PWR nuclear production units rated at 1300 MW. The first two units were built and commissioned in 1985 and 1986.

Complementary investigations were carried out in the early 90s in order to set up on the site, two additional units rated at 1400MW (type N4) instead of two units rated at 1300 MW as initially planned. These studies found that the project was feasible in every area, including that of the environment. EPR which offers a significant improvement in terms of the environment with respect to the N4 units does not therefore pose any particular problems.

The geographical location of Flamanville is a definite advantage in environmental terms: The very strong marine currents prevailing in this area of the Cotentin would offer a fast and efficient method of donating the thermal output from the plant.

| Seq. No 174 | Country | Article Article 17 | Ref. in National Report |
|---|---|---|---|

Question/ Comment: The utilisation of existing NPP sites is very probable approach also in other countries which relay on future renesation of nuclear power Can you, please describe in more details the process of relicensing (confirmation) of the site Flamanville which was selected for new NPP with EPR?

Answer: In France, the process of relicensing of a nuclear site is not different from the licensing of a new one (see report p. 26 & pp. 107-108).

First, a national public debate about siting takes place, dealing with all aspects of the project (socio-economics, environmental aspects, etc…). The ASN has no specific role during this debate, which is organised by the National Commission for Public Debate (set-up in case of decision about large investments).

Then the processing of the plant authorisation application includes a local public inquiry, based on a hazard analysis and an environmental impact assessment provided to the public.

In the specific case of the relicensing of a site, the environmental impact assessment is made by taking into account both the new nuclear installation itself and all the onsite operating nuclear installations, with updated environmental data.

Meanwhile, based on a Preliminary Safety Report, ASN naturally assesses the plant design according to the safety-related characteristics of the site (seismicity, hydrogeology, industrial environment etc…).

| Seq. No 175 | Country | Article Article 17 | Ref. in National Report |
|---|---|---|---|

Question/ What are the regulatory procedures for survey and evaluation of capable fault or

Comment geological structure suspicious of a capable fault without evidences, found at or near the site area of nuclear facilities in operation or under licensing review process? If there are nuclear facility sites that were (or are) engaged in this procedure, what were(are) the sites and how were(are) the issues resolved?

Answer The French regulatory practice does not envisage renewal of licence. This practice can however be comparable with safety review. Within the framework of the periodic safety review required by ASN every 10 years, the seismic revaluation particularly consisted in applying the basic safety rule RFS 2001-01 on seismic risk determination. This rule envisages in the case of a site located in the immediate vicinity of an active fault with a rupture surface, a study aiming at determining the seismic movements associated with the seism having been able to occur on this fault, and being able to have an effect on the site. However, no French PWR was built on a fault of active surface able to generate displacements on the ground in the event of seism.

**Art. 18 – Design and construction**

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 176 | | Article 18 | § 2.3.9+§18.2.1p109 |

Question/ Comment: What is the status of the licensing process for the proposed French EPR and to what extent will the ASN 2003 and 2004 review of project safety studies be given credit in the licensing process?

Answer: In France, well before submitting an authorization application, an operator is expected to submit the safety options of its project in an unofficial though usual preparatory procedure which enables the regulator to influence the design at an early stage of the project.

As a conclusion to such an assessment, the French Nuclear Safety Authority took position at the end of 2004 on the acceptability of the EPR safety options proposed, and informed the operator of their technical recommendations as well as the issues that will have to be taken into account in the authorization decree application.

This public position can then be considered as equivalent to a generic "preliminary design certification" delivered by some foreign Nuclear Safety Authorities.

Currently, the preparatory work still goes on with the assessment of a draft Preliminary Safety Analysis Report, and some detailed studies on which it is based on.

Such detailed preparatory assessment is expected afterwards to facilitate, but in no way replace, the authorization decree application phase, which is expected to take place at the beginning of 2006.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 177 | | Article 18 | |

Question/ Comment: Please provide an outline of the safety analyses conducted and the results obtained for design changes and NPP upgrading measures?

Answer: For existing reactors, see the answer to question No 88 under Article 14.

The assessment of EPR project with the Nuclear Safety Authority led to a large number of safety analyses conducted and some of the results obtained led to design changes. Here are some examples:

- Improvements in the design of the Fuel Pool Cooling System:
Following the FCPS safety evaluation of year 2000, EDF proposed a new design of the Fuel Pool Cooling System with a supplementary train (3rd train) to respond to the Nuclear Safety Authority recommendation and to achieve the following two goals:

- to drastically reduce the risk of boiling in the fuel pool ;
- to “practically eliminate” the risk of fuel melting in the pool.

This supplementary train allows the risk of boiling to be cut down in case of the loss of the two main FPCS trains- RRCA event- , especially in case of:

- total loss of cooling chain (TLOCC): the third train is cooled by a dedicated intermediate cooling chain which is independent of the CCWS system
- or in case of complete loss of ultimate heat sink (LUHS) : the third train is

cooled by a diversified ultimate heat sink which is common to the CHRS system and the unavailability value of which is estimated to correspond to 10% of the frequency of total loss of the ultimate heat sink (LUHS).
- or in case of station blackout (SBO). The third train is backed by an SBO-diesel generator in states D to F in case of SBO.

The probability assessment of this design shows that the risk of boiling in the fuel pool is reduced by a factor 15 compared with the design presented in Basic Design Report 1999 and that the risk of reaching unacceptable criteria is reduced by a factor 40.The frequency of boiling in the pool is estimated at 1.5x10-5 /r.y. and the frequency of reaching unacceptable conditions at 7.6x10-9 /r.y. The risk of melting of the fuel in the pool is thus considered practically eliminated.

PSA Studies

PSA level-1 (N1):

The PSA studies of EPR carried out during 2002/2003 enabled the probability evaluation presented in the Basic Design Report (BDR) 1999 to be completed. These studies ensured that the design was in compliance with the general safety objectives.

The main developments taken into account from the BDR 1999 concern: operating profile, preventive maintenance during power operation, equipment reliability data in accordance with the German, French and international OEX, modelling of I&C chains into “Compact Model”.

The results obtained and incorporated in the PSAR edition 2003 confirm that the EPR Project level 1 probability objectives are respected.
- The overall frequency of the Core melting is estimated at: 5.4 10-7/r.y
- The overall frequency of the Core melting corresponding to the reactor power states is lower than 10-6/r.y: (5 10-7/r.y)
- The overall frequency of the Core melting during shutdown states is lower than that of states in power: (4.1 10-8/r.y)

PSA level-1+ (N1+):

Grouping together the core melting sequences identified in PSA N1, in accordance with the damage state on confinement in excess of core melting, leads to the following results incorporated in PSAR edition 2003 which confirms that the Project probability criteria are respected:
- The sum of PDS 2 (melting of the core with potential later failure of the housing (with non-availability of EVU/CHRS)) and PDS 3 (Melting of the core with potential early failure of the housing) is less than 10-6/r.y (6.9 x 10-8/r.y)
- The value of PDS 3 (melting of the core with early loss of confinement) is less than 10-7/r.y (3.17 x 10-8/r.y)

List of RCCA events

In addition to the incidental and accidental reference transients; operating conditions with multiple failure are considered as part of the safety demonstration. A preliminary list of the situations to be taken into account, derived from experience with the French operating plant, was proposed in BDR 99 based on the available studies.

In compliance with the request from Technical Guidelines, EDF proposed a revision to that list for the PSAR, taking into account the results of the latest EPR

PSA level 1.
The methodology used for EPR is consistent with that approved by the ASN on the existing plants to determine the list of supplementary situations. It uses the results of the PSA N1 studies carried out at conception to detail the specific RRCA provisions, the contribution of which is found to be necessary to reduce the frequency of core melting.
The updated list presented in the PSAR edition 2003 includes 11 operating sequences.
Compared to the preliminary list issued during BDR 99, certain operating sequences which were found to be negligible from a probability viewpoint (CMF between 10-9 and 10-11) were cancelled whereas three sequences corresponding to CC or operator failures which are found to be important from a probability point of view have been added.
The corresponding support studies are presented in the PSAR; the list will be re-evaluated if necessary to be consistent with developments in the EPS studies at the time of the Provisional Safety Analysis Report.

Confinement containment:
The confinement containment of the EPR Project proposed during the Basic Design was a double wall containment concept designed to withstand all dimensional situations including severe accidents. It consisted of an outer wall in reinforced concrete and an internal wall in prestressed concrete on which the addition of a composite liner was envisaged to improve leaktightness beyond dimensional pressure; The studies carried out by EDF concerning the life-time of the 900MW containment and the REX behaviour of the double wall containment on French operating plants, to which are added the difficulties of guaranteeing that composite systems will have a life-time of 60 years, led EDF to propose a solution of a double wall with a metal liner on the internal containment.
The adoption of this new design made it possible to separate, within the confinement function, the aspect concerning leaktightness provided by the metal liner from the aspect concerning pressure resistance provided by the prestressed concrete in the internal containment.

The dimensional pressure values chosen in the design of the containment are as follows:

- An absolute dimensional pressure fixed at 5.5 Bars which covers all the dimensional scenarios including the multiple failure scenarios with the core melting ( RRCB).
- A maximum test pressure value set at 6 bars: taking particular account of the effects of temperature on the liner in an accidental situation.
- As concerns the defence in-depth and with the purpose of ensuring the existence of (safety) margins, taking account of an absolute checking pressure set at 6.5 bars which enables the leaktightness of the internal containment to be maintained in the event of serious accident limit scenarios, taking into account aggravated hydrogen production phenomena.

From the leaktightness point of view, EPR adopts a maximum internal containment leak rate identical to that set under the terms of the 900 MWe licence, that is to say 0.3% vol/day.
Collecting leaks is based on the following principles:

- All the containment penetration end at the peripheral buildings which collect leaks by suitable means

- Potential leaks in the liner are collected in the space between the inner and outer walls and are processed by a safety classified ventilation system .

This new design was analysed by ASN in the framework of Advisory Committee of experts for nuclear reactors at mid-2004 and approved by the DGSNR in the safety options letter in September 2004

| Seq. No 178 | Country | Article Article 18 | Ref. in National Report |
|---|---|---|---|

Question/ Comment: How have the consequences in case of civil aircraft impact been considered in safety analyses for the French NPPs?

Answer: This issue is addressed in specific studies, which are also related to security and physical protection measures.
By nature the security measures are not included in the scope of this Convention and, in compliance to French law, any information related to such measures cannot be disseminated to the outside without a confidential agreement between two Governments. Such agreements exist between France and some Contracting Parties, to which it should be referred for further information on the topic.

| Seq. No 179 | Country | Article Article 18 | Ref. in National Report |
|---|---|---|---|

Question/ Comment: Have special measures in design safety been taken to minimize the consequences of civil aircraft impact on NPPs?

Answer: This issue is addressed in specific studies, which are also related to security and physical protection measures.
By nature the security measures are not included in the scope of this Convention and, in compliance to French law, any information related to such measures cannot be disseminated to the outside without a confidential agreement between two Governments. Such agreements exist between France and some Contracting Parties, to which it should be referred for further information on the topic.

| Seq. No 180 | Country | Article Article 18 | Ref. in National Report |
|---|---|---|---|

Question/ Comment: Please explain how do you demonstrate (prove) the second level of defence-in-depth concept at PWRs operated in France (control of abnormal operation and detection of failures).

Answer: Several means participate to the second level of defence in depth ; for instance :

- detection systems (leaks, vibrations, temperature,…)
- protection systems (to avoid some parameters exceed specified values)
- periodic tests
- in service inspection
- conformity check (part of PSR)

For each of these means, justification is provided, for instance transient calculations to verify the reactor protection system, demonstration of the conformity to regulations, study of the comprehensiveness of the periodic tests,…

| Seq. No<br>181 | Country | Article<br>Article 18 | Ref. in National Report |
|---|---|---|---|

Question/ Comment — Are any investigations (research) made on fire and fuel explosion from an aircraft as a consequence of its interaction with the NPPs?

Answer — This issue is addressed in specific studies, which are also related to security and physical protection measures.
By nature the security measures are not included in the scope of this Convention and, in compliance to French law, any information related to such measures cannot be disseminated to the outside without a confidential agreement between two Governments. Such agreements exist between France and some Contracting Parties, to which it should be referred for further information on the topic.

| Seq. No<br>182 | Country | Article<br>Article 18 | Ref. in National Report<br>2.3.9,P.11 |
|---|---|---|---|

Question/ Comment — In Section 2.3.9, it is mentioned that “in 2003 and 2004 the ASN continued its review of the project safety studies, in particular relying on the Advisory Committee for nuclear reactors and on foreign experts”.
More information would be appreciated on the project safety studies.

Answer — Same Answer as to question N° 177:
The assessment of EPR project with the Nuclear Safety Authority led to a large number of safety analyses conducted and some of the results obtained led to design changes. Here are some examples:

- Improvements in the design of the Fuel Pool Cooling System:
Following the FCPS safety evaluation of year 2000, EDF proposed a new design of the Fuel Pool Cooling System with a supplementary train (3rd train) to respond to the Nuclear Safety Authority recommendation and to achieve the following two goals:

- to drastically reduce the risk of boiling in the fuel pool ;
- to “practically eliminate” the risk of fuel melting in the pool.

This supplementary train allows the risk of boiling to be cut down in case of the loss of the two main FPCS trains- RRCA event- , especially in case of:

- total loss of cooling chain (TLOCC): the third train is cooled by a dedicated intermediate cooling chain which is independent of the CCWS system
- or in case of complete loss of ultimate heat sink (LUHS) : the third train is cooled by a diversified ultimate heat sink which is common to the CHRS system and the unavailability value of which is estimated to correspond to 10% of the frequency of total loss of the ultimate heat sink (LUHS).
- or in case of station blackout (SBO). The third train is backed by an SBO-diesel generator in states D to F in case of SBO.

The probability assessment of this design shows that the risk of boiling in the fuel pool is reduced by a factor 15 compared with the design presented in Basic Design Report 1999 and that the risk of reaching unacceptable criteria is reduced by a factor 40.The frequency of boiling in the pool is estimated at 1.5x10-5 /r.y. and the frequency of reaching unacceptable conditions at 7.6x10-9 /r.y. The risk of melting of the fuel in the pool is thus considered practically eliminated.

PSA Studies

PSA level-1 (N1):

The PSA studies of EPR carried out during 2002/2003 enabled the probability evaluation presented in the Basic Design Report (BDR) 1999 to be completed. These studies ensured that the design was in compliance with the general safety objectives.

The main developments taken into account from the BDR 1999 concern: operating profile, preventive maintenance during power operation, equipment reliability data in accordance with the German, French and international OEX, modelling of I&C chains into "Compact Model"

The results obtained and incorporated in the PSAR edition 2003 confirm that the EPR Project level 1 probability objectives are respected.

- The overall frequency of the Core melting is estimated at: 5.4 10-7/r.y
- The overall frequency of the Core melting corresponding to the reactor power states is lower than 10-6/r.y: (5 10-7/r.y)
- The overall frequency of the Core melting during shutdown states is lower than that of states in power: (4.1 10-8/r.y).

PSA level-1+ (N1+):

Grouping together the core melting sequences identified in PSA N1, in accordance with the damage state on confinement in excess of core melting, leads to the following results incorporated in PSAR edition 2003 which confirms that the Project probability criteria are respected:

- The sum of PDS 2 (melting of the core with potential later failure of the housing (with non-availability of EVU/CHRS)) and PDS 3 (Melting of the core with potential early failure of the housing) is less than 10-6/r.y (6.9 x 10-8/r.y)
- The value of PDS 3 (melting of the core with early loss of confinement) is less than 10-7/r.y (3.17 x 10-8/r.y).

List of RCCA events

In addition to the incidental and accidental reference transients; operating conditions with multiple failure are considered as part of the safety demonstration. A preliminary list of the situations to be taken into account, derived from experience with the French operating plant, was proposed in BDR 99 based on the available studies.

In compliance with the request from Technical Guidelines, EDF proposed a revision to that list for the PSAR, taking into account the results of the latest EPR PSA level 1.

The methodology used for EPR is consistent with that approved by the ASN on the existing plants to determine the list of supplementary situations. It uses the results of the PSA N1 studies carried out at conception to detail the specific RRCA provisions, the contribution of which is found to be necessary to reduce the frequency of core melting.

The updated list presented in the PSAR edition 2003 includes 11 operating sequences.

Compared to the preliminary list issued during BDR 99, certain operating sequences which were found to be negligible from a probability viewpoint (CMF between 10-9 and 10-11) were cancelled whereas three sequences corresponding to CC or operator failures which are found to be important from a probability point of

view have been added.
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This new design was analysed by ASN in the framework of Advisory Committee of experts for nuclear reactors at mid-2004 and approved by the DGSNR in the safety options letter in September 2004.

| Seq. No | Country | Article | Ref. In National Report |
|---|---|---|---|
| 183 | | Article 18 | 2.3.9,P.11 |

Question/ In Section 2.3.9, it is mentioned that there is an improved cooperation between the

Comment Nuclear Safety Authorities.

More information would be appreciated on the cooperation between the Nuclear Safety Authorities.

Answer Concerning the cooperation between STUK and DGSNR on the EPR project :

1) DGSNR gave STUK a copy :

- of all the technical reports presented by IRSN to the French Advisory Committee for Nuclear Reactors since 1994 ;
- of the “Technical Guidelines for the design and construction of the next generation of nuclear power plants with pressurized water reactors”, adopted by the French Advisory Committee for Nuclear Reactors in October 2000 at the end of the Basic Design Review Phase, and endorsed by DGSNR.

2) a Finnish expert from STUK has been nominated as a member of the French Advisory Committee for Nuclear Reactors ;

3) semi-annual meetings have been held to share technical information on each other assessment topics in progress

4) DGSNR and STUK shared information on the control of the design and construction studies of components of the primary and secondary circuits.

**Art. 19 - Operation**

| Seq. No | Country | Article | Ref. In National Report |
|---|---|---|---|
| 184 | | Article 19.2 | 19.2.2, page 119 |
| Question/ Comment | Could you describe how and when the main system configuration control is performed after Unit start up? | | |

Answer

In order to get a better understanding of the surveillance organisation, let us remain that to reach and maintain a safety level, three steps are necessary.

1. At design stage, a safety reference system is defined. The designer attributes roles to each back up, and more generally safety related systems, and defines associated performances.
2. The assurance that the design safety reference system has been met is gained during commissioning, through factory controls, in process installations checks, qualification tests and preoperational test.
3. During operation a surveillance process is developed in order to ensure that the safety reference system does not change.

The surveillance during operation may be sorted into two classes.

- An item or system has to be repaired, the installation can be changed and the performances of the new or repaired item may be slightly different. Re-qualifications tests are required in order to make sure that the reference is still met.
- Without any breakdown or failure, normal operation surveillance should be performed :
    - o Test associated to preventive maintenance,
    - o In service installation checks,
    - o GOR's periodic tests (1)
    - o Valves position checks....

(1) GOR = General Operating Rules (GOR specify all the safety requirements governing the unit operation.)

The GOR's periodic tests aim at ensuring that there is no negative evolution of the design safety reference system, that the unit complies with the hypothesis of the "post accident " studies, that back up systems and safety functions are operable, that post accident procedures will be operable. In order to develop the different surveillance needs, the organisation is the following:

During outage of the unit for refuelling, maintenance is performed; some important items are dismantled for control. Re-qualification tests are performed. A methodology was drawn up in order to point out the proper control related to the type of maintenance or content of the repair.
According to Technical Specifications, the different states of a PWR unit are sorted into 6 "operational domains". For each domain, the safety problems are similar so that the TS's requirements are the same. When starting up, a stop point is programmed at each "domain" change in order to control whether the required operability of safety functions for the coming domain is met or not. During those stop points, alignments are checked.

During start up, periodic test are performed following a pre-established schedule and when the unit status allows their performance.

The results of those different tests are collected and submitted to a special committee, which, actually, includes Nuclear Safety Authority. At the end of the process a formal authorization for Power operation is given.

During power operation, periodic tests are still performed (every month or two months for most of them) in order to compare the results of measures to criteria so that the operability of safety related system could be assessed.

| Seq. No 185 | Country | Article Article 19.2 | Ref. in National Report Section 19.2.4 |
|---|---|---|---|

Question/ Comment

Section 19.2.4 mentions two types of emergency procedures: event-oriented procedures and state-oriented procedures. Besides, it is noted that the actions that have to be taken in case of an incident or accident are described in Chapter 6 of the General Operating Rules. The last paragraph of this section mentions the types of incidents and accidents covered by state-oriented emergency procedures.
1) Do you have at nuclear plants both event-oriented and state-oriented emergency procedures?
2) What actions are specified in Chapter 6 of the General Operating Rules and how do they correlate with emergency procedures?
3) Are there specific procedures that define personnel actions during severe accidents?
4) Why do the state-oriented emergency procedures fail to cover failures and events involving reactivity oscillations?

Answer

Q1):
No, we have never got both event-oriented and state-oriented emergency procedures at the same NPP. On each nuclear power station, there are either event-oriented procedures or state-oriented procedures.

Q2):
No actions are specified in chapter 6 of the general operating rules. Operating actions are specified in emergency procedures, which belong to this chapter 6. In chapter 6 there are emergency procedures, in chapter 9 there are periodic test procedures, etc... Operators use, in main control room, emergency procedures where they can find emergency operating actions.

Q3):
Yes, there is a specific procedure to define personnel actions during severe accidents : the "SAMG" Severe Accident Management Guideline.

Q4):
Normal operating procedures allow avoiding reactivity oscillations. More over if such event occurs, reactor protections will become involved (automatic shutdown/reactor trip) and so reactivity oscillations will be over.

| Seq. No 186 | Country | Article Article 19.3 | Ref. in National Report |
|---|---|---|---|

Question/ Comment

Since 1980s, problems related to sizing or control switch setting of safety-related motor-operated valve in nuclear power plants have been identified and programs have been established for solving these problems. For example, United States issued Generic Letter 89-10(Safety-Related Motor-Operated Valve Testing and

Surveillance) and 96-05 (Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves) to solve these problems.
Did you experience similar problems in MOV? Is there any plan to cope with the problems about safety-related motor-operated valve? If yes, please explain the plan briefly.

Answer The verification of the sizing of motorisation of motor-operated valve has been performed as a part of the conformity check associated to the 20-years PSR. The extent of this verification was limited to the design situations, and the corrective actions were introduced in the batch of modifications associated to the second 10-years outage. An extension of this verification is now being performed for some circuits, which have potentially to operate in severe accidents situation.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 187 | | Article 19.3 | page 128 |

Question/ Comment Please explain, how the maintenance method OMF („Maintenance optimization by reliability") works in practice and give some short examples of implementation of this method. (section 19.4.1.1.5)

Answer The OMF method comes from the conversion by EDF of a US method known as RCM (Reliability Centered Maintenance), used in the American civil and military aeronautical field in the 70's, to control the operating cost of aircraft in the increasingly competitive context of that industry, while maintaining or improving flight safety.
The Nuclear Safety Authority (ASN) is currently examining this maintenance method.

At EDF, two essential objectives were targeted by the implementation of OMF:

- simultaneously gaining in several company challenges:
    - in safety by better monitoring of the equipment considered to be the most sensitive,
    - in competitiveness, by reducing failures leading to production losses and by eradicating pointless maintenance operations,
    - in control of plant unit outages periods and the quality of services by limiting the number of tasks as a consequence of their improved justification.
- obtaining an initial level of systematic preventive maintenance program optimization, an approach to reduce maintenance volumes which backs up the system by implementing conditional maintenance and sampling by reference devices.

The goal of the OMF method is to identify the equipment, and even the components, for which failures or degradation considered to be unacceptable are likely to occur, then define ad hoc prevention arrangements. Implementation is possible as soon as the device allows the collection and analysis of experience feedback and is set up in place.
The OMF method was implemented to define the maintenance programs for the 50 most important systems out of the 58 PWRs forming the EDF Nuclear Inventory, and out of the 20 main Fuel and Carbon Thermal Inventory systems operated by EDF.

In a simplified manner, the OMF approach breaks down into three major phases:

Phase 1 – Evaluation of stakes /A

The goal is to identify the equipment and even the components for which unacceptable failures or degradations are likely to occur and with respect to which avoidance action should be established.

Functional analysis at system level is a way of:

- identifying the functions of the system and equipment contributing to covering these functions,
- identifying pertinent failure and degradation modes of this equipment,
- determining the equipment that is a "serious" with regard to safety, environment, availability and maintenance cost challenges.

For each of the challenges taken into consideration, seriousness is evaluated on a binary basis (a failure is serious or is not) on the basis of predefined criteria:

- Safety: impact on the Technical Operating Specifications, impact on core meltdown risks, entry into accidental procedures;
- Environment: confirmed or potential risk of rejection of products reputed to be harmful,
- Availability: total or partial loss of production, loss of efficiency, lengthened shutdown times;
- Maintenance: failure directly or indirectly generating excessive expenses,
- Non-serious aspect not addressing the previous criteria and that are corporate choices.

A synthesis table, known as FMECA (Failure Mode Effect and Criticality Analysis), evaluates for each piece of equipment the possible causes and modes of failure associated with the nature of their seriousness regarding the consequences. For instance, for the EAS (Containement Spray System) system spraying water into the containment enclosure in an accidental phase, a failure identified as "inadvertent stoppage" is considered to be "safety-serious" with respect to the "EAS pump" equipment because it will result in the failure of the system assignments.

Phase 2 – Evaluation of stakes /B

For each piece of equipment, the second stage of analysis consists in locating the envisioned cause of failure and its frequency of occurrence by examining experience feedback (in some cases, expert judgment may prove to be necessary). For instance, in the case of the aforementioned "EAS pump" examination of experience feedback, for the chosen period, reveals a definite case of a "Bearing-stator high temperature" alarm related to a lubricating defect. This event is taken into consideration as a degradation that may cause the inadvertent stoppage of the pump.

Experience feedback plays a fundamental part in the OMF method. Indeed, research into serious failures during phase 1, but the choice of maintenance tasks in phase 3 also, requires thorough knowledge of the degradation mechanisms that may result in failure. For this purpose, EDF uses a computer tool (SAPHIR), filled in at the source by the local maintenance people and systematically comprising data the nature of which is specified nationally by the central services. Failures and degradations can also be described and counted for each piece of equipment and data for understanding the reliability and cost of maintenance are filled in.

Phase 3 – Optimisation of maintenance

If the above recommendations are taken into consideration, maintenance tasks can be created, or existing tasks be modified, to construct or realize the Preventive Maintenance Programs (PMP) by adapting maintenance tasks to the identified causes of failure and degradation.
OMF does not directly include the method for defining the type of maintenance action needed, confined to the choice of a strategy according to a set of possible situations. The choice of maintenance tasks results from knowledge of the industrial methods used to deal with failure causes to be avoided. Accordingly for EDF, some PMP's propose for the same result, alternative preventive maintenance actions (intrusive or conditional maintenance for incident) with the final choice being made at local level by direct players depending on their proficiencies.
To complete the example of the EAS pump: The "inadvertent stoppage" failure mode is therefore "safety-serious" with respect to the "EAS pump" equipment and with respect to experience it will be necessary to establish preventive maintenance actions of the "periodic lubricating" type to forestall it.

Safety complement: use of EPS
Read here also the answer to Question No 112:

PSAs are used in France to supplement the conventional deterministic analyses. They are considered by ASN as an interesting tool in the definition and prioritisation of the actions to be taken in order to attain or maintain a satisfactory safety level. Their main applications for French NPPs include the following safety areas:

- Periodic safety review,
- Probabilistic event analysis,
- Design of future reactors,
- Importance of systems and equipment with regard to safety,
- Operational technical specifications.

Since 1990, a level-1 PSA has been developed in France which covers now all internally initiated events except aggressions, including all applicable reactor states including shutdown. In 2004, the scope of level-1 PSA performed by IRSN for 900 MW plants was extended to include an internal aggression such as fire. In the same year, as mentioned above, the utility developed a level-2 PSA for 900 MW plants covering all applicable reactor states including shutdown.

PSA level 1 have been developed for all EDF NPP series and for the French EPR Project for power and shutdown states. A level 2 PSA has been achieved for EDF 900 MW series and is in preparation for the 4-loop EDF NPP (1300 and N4). A level 2 PSA is foreseen for the French EPR Project.
Integrated PSA models are used to evaluate the safety of French NPP during periodic Safety Review.

The risk profile is balanced. The main contributors for level 1 PSA are the loss of 6,6 kV safeguard switch board by common cause failure and failures of reactor cooling pump and transients without reactor trip caused by rod blockage.

The main contributors for level 2 PSA are heterogonous dilutions with large early releases and situation with basemat melt through with late releases.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 188 | | Article 19.3 | A19.3.3 P125 |

Question/ Comment: It is written that "operating rules to be followed in the case of incident or accident situation are described in Section 6 of the General Operating Rules and Procedures for hypothetical and last resort situations are described in its Section 10. They are approved by the Nuclear Safety Authority". Normally the regulatory bodies approve the higher level documents such as final safety analysis report, plant technical specifications including emergency plans, however, the procedures are the working level documents which are prepared and approved by the licensee management. These documents are prepared for working within the approved domain of the regulatory body, since the licensee has the ultimate responsibility for safety. By approving the procedures such as incident & accident procedures, by the ASN, it seems that the regulator is sharing the responsibility for safety (legal liability) with the licensee. France may kindly clarify?

Answer: The general operating rules and the technical specifications are sent before their use to the Nuclear Safety Authority for approbation. The general rules for incidental and accidental approach are included in the General operating rules. They are transcribed in incidental and accidental operational procedures, which are not sent to the Nuclear Safety Authority, but stay under responsibility of the licensee. Their contents may be examined during specific inspections.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 189 | | Article 19.4 | |

Question/ Comment: What is the status of the implementation of the SAMGS (or equivalent procedures) at French NPPs?

Answer: The French SAMGs are called GIAG, as Guide d'Intervention en Accident Grave. They are to be applied in all French PWR. They are obligatory and are classified as class 4 documents by EDF directive DI 001

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 190 | | Article 19.4 | p. 126, 19.4.1 |

Question/ Comment: Which countermeasures have been taken for the time until major modifications are implemented to restore the safety margins in case of sump filter clogging?

Answer: Same answer as to question N° 123:
The durations allowed by ASN for treatment of non-conformances are balanced according to the impact on the installations safety, to the probability of occurrence of the initiator likely to generate the defect, and to the feasibility of repairs. The implementation of palliative provisions, in material term (temporary modification), or organisational (control, maintenance, surveillance) may allow to accept longer times. In the case of the anomaly relating to the risk of the sumps filters clogging in accidental conditions, the transitional provisions retained by the utility EDF concern:

- Opening of the pre-filters doors of the 900 MWe plant units;
- Avoidance of powdery heat insulation installation (microtherm) in the Building Reactor;
- Possibility of supplying in water the tank of the reactor cavity and spent fuel pit

cooling and treatment system.

As an example, interim measures were recently adopted by EDF to reduce the risks associated with the "sump clogging" problem generic to PWRs, during the time needed to implement the final corrective measures.

With respect to the operation of the Safety Injection System (RIS)/ Containment Spray System (EAS) with sump in recirculation mode, a number of measures have been taken or analyzed to minimize the generation of debris or the consequences of sump fouling and its impact on the recirculation function. A possible answer could be:

Opening of the prefilter doors of the 900 MWe plant unit.
In this plant unit, the RIS/EAS sumps are equipped with large mesh pre-filters a few meters upstream. In the case of these filters becoming fouled, in certain circumstances, a damming phenomenon may occur, resulting in the filters becoming clear. In this event, there is a risk of air being drawn in at the Safety Injection System and Containment Spray System pumps. Accordingly, a provisional arrangement (EDF-DPN-DT n°192) was issued in early 2004 in order to have the prefilter doors open and eliminate any risks of dams forming.

Avoidance of powdery heat insulation installation (microtherm).
Powdery heat insulation debris considerably increases the load losses from filter debris mattresses. EDF took the decision in early 2004, insofar as possible, not to install this type of heat insulation inside the Reactor building during operational or maintenance work.
Further, a decision was reached at the same time to minimize the quantity of this type of heat insulation during the future replacement of the SGs. Accordingly, this provision, applied to the replacement of the SGs in plant unit 4 at Tricastin has made it possible to reduce the amount of powdery heat insulation by a factor of 20 or so compared to the initial predictions.

Optimization of control procedures.
The control procedures were analyzed to identify any potential optimization sources with respect to fouling risks. For the "short term" aspect before the fitting out of the emergency teams, the current procedures already appear to be satisfactory and it will be necessary to ensure that the change of control does not degrade the overall safety level of the installations. It is also noteworthy that the limitation of the Safety Injection System and Containement Spray System flow-rates already appears in the procedures (shutdown of a Containment Spray System line, shutdown of Safety Injection System pumps or realignment…) when so permitted by the parameters. Therefore, it is unadvisable to go any further in the short term flow rate reduction.

Reactor Building cleanliness.
On the basis of the survey of the site methods used, it has been verified that the reactor building cleanliness was satisfactory. The data obtained from this survey was added to the reference material for the purpose of sizing hypotheses.
Other avenues have been explored, for instance the design of deflectors to trap debris upstream of the filters, but in the same way as for the control procedures, these avenues were not deemed to be pertinent and did not lead to any concrete actions.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 191 | | Article 19.4 | 19.2.2,page 117 |

Question/ Comment: Which way the main actions are being used for validation of design and beyond design basis accidents? Are there any differences between procedures for validation of design basis accidents and validation of beyond design basis accidents?

Answer: For the design basis accidents, no operator action is accounted during a minimal period (e.g. 20 minutes for actions in the main control room) corresponding to the state diagnosis and to the performance of operator actions. After this period, actions are supposed to be performed in accordance with emergency operation guidelines, without any mistake, as soon as the conditions for these actions are met. For the design extension accidents (corresponding to events combination determined by PSA), necessary operator actions (for meeting safety criteria) are delayed to the maximum acceptable period, in order to provide the maximum acceptable delay to perform these actions. Other important actions are performed when their probability of failure becomes residual.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 192 | | Article 19.4 | 19.2.4, page 121 |

Question/ Comment: In the first stages of emergency accident response actions are based on the event-oriented procedures, how do you transfer from event-oriented procedures (especially if the accident complication takes place) to state-oriented procedures?

Answer: For each nuclear power station in France, there are either event-oriented procedures or state-oriented procedures, which are implemented but never both. So there is never transfer from event-oriented procedures to state-oriented procedures in NPPs.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 193 | | Article 19.4 | 19.2.4, page 121 |

Question/ Comment: Are there any areas or situations where or when the state-oriented procedures are being used for severe accidents?

Answer: For severe accident situation, a specific procedure is used , the "SAMG" : Severe Accident Management Guideline. State-oriented procedures are not used any more in situations of severe accidents. Specific physical criteria are used for the tranfer from state-oriented procedures to "SAMG". Both sets of procedures are never used in the same time (either State oriented procedures or SAMG).

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 194 | | Article 19.4 | 19.4.1.3.2, page 129 |

Question/ Comment: Would you describe how the personnel actions at severe accidents were validated?

Answer: 1. If "personnel actions" means explaining how will we initially validate the content of operating action is required in the SAMG : this derives from supporting studies, experts, the CEA and even from a number of tests. These actions have not been "validated", strictly speaking, but they are based on physical considerations, common sense, and actions that have already been requested(and validated) by

State-oriented approach.

2. Or whether it is a matter of validating operator actions in real time, requiring prior agreement of this action by crisis teams : this is done through proprietary procedures which cannot be described here.

| Seq. No 195 | Country | Article<br>Article 19.4 | Ref. in National Report |
|---|---|---|---|

Question/ Comment: Please provide info on what is the actual status of the SAMGs (severe accident management guidelines) implementation in NPPs. Are they obligatory for all operating NPPs in France?

Answer: Same Answer as to question N° 189:
The French SAMGs are called GIAG, as Guide d'Intervention en Accident Grave. They are to be applied in all French PWR. They are obligatory and are classified as class 4 documents by EDF directive DI 001

| Seq. No 196 | Country | Article<br>Article 19.4 | Ref. in National Report |
|---|---|---|---|

Question/ Comment: What are the main mitigative and preventive measures considered in the SAMGs (severe accident management guidelines) at PWRs? Do you consider an external reactor pressure vessel cooling to
retain the molten core inside the vessel?

Answer: 1/ The SAMGs at PWRs consider immediate and delayed actions.
Main immediate actions are opening of all pressuriser relief valves to evacuate energy out of reactor vessel and avoid vessel failure at high pressure, checking of containment isolation valves in the closed position in order to limit the release in environment, confirming safety injection at maximum flow to fill reactor vessel and evacuate energy, cooling at maximum rate by non radioactive steam generators to evacuate energy outside containment and confirming containment spray system is on to limit containment pressure.
Main delayed actions are: injecting high flow borated water in reactor vessel to cool the core or keep molten core in vessel, running containment spray system to limit containment pressure if there is no hydrogen risk by desinerting containment, opening of long term containment depressurisation system if pressure is too high in the containment.
There is no system dedicated for external reactor pressure vessel cooling at PWRs, so this is not considered in SAMG as a mitigative measure. Anyway, water from containment spray system leading to reactor cavity, would participate to reactor vessel cooling.

2/ MITIGATION: Allowance for certain severe accidents

The safety approach is supplemented by allowance for a nuclear steam supply system state characterized by core degradation, with the primary system depressurized (operating conditions designated "non-plausible"). The principle of the approach is to have available the necessary resources to delay and limit the consequences of events leading to such a state (maintaining, adequate confinement for a sufficiently long period), to enable *the* public authorities to take off-site protective action (application of the Off-site Emergency Response Plan which

notably. includes the evacuation of people within a 5 km radius and sheltering within a 10 km radius, within 12 to 24 hours).
Introducing guidelines and procedures calling for available conventional resources and special “last- resort” resources carries this out:

- U4 provisions which avoid early contact between corium and the environment if the former attacks the reactor building basemat,
- U5 provisions for depressurizing and filtering the containment atmosphere, thus ensuring that the integrity of the third barrier is maintained,
- SPE procedure (SPE is a French acronym for state-oriented surveillance) and U2 procedure enabling the operating team to carry out special surveillance of the containment and take the necessary action to restore the confinement function, in addition to priority core safeguard action,
- the Severe Accident Response Guideline which details the special action to safeguard the core and, as a priority, ensure optimal confinement of the radioactive substance, for as long as possible, and after having activated at an early stage the National Emergency Response System to manage the situation.

The support documents are formally laid down in the Emergency Response Team Action Guideline which provide additional material to that in the operating procedures applicable to plausible operating conditions and which it takes over from in the long term.

These special resources and guidelines are associated with an objective of the release from the containment, which is referred to as "S3 source term", which is compatible with the Off-site Emergency Response Plan. This S3 source term, which is representative of delayed releases via pathways leading to certain retention of the main radionuclides, is less than 1% of the core inventory (75% for the noble gases).

The installation of hydrogen recombiners is planned, provided their effectiveness and harmlessness is demonstrated. These recombiners would reduce the quantity of hydrogen liable to deflagrate (production of hydrogen in the reactor vessel essentially results from oxidation of the fuel cladding, and outside the reactor vessel of oxidation of other metals by steam, particularly in the event of attack of the basemat by corium).

It is to be noted that allowance for severe accidents closely links the "design" and "operation" aspects.

A- SEVERE ACCIDENT PROVISIONS

The S1 source term (representative of accidents with early loss of containment) corresponds to high-energy accidents that are of extremely low probability and therefore for which it is not necessary to make special provisions.

As the potential consequences of the S2 source term (representative of accidents involving release from the containment directly into the atmosphere after delayed loss of leaktightness: at least one day after the accident) can be difficult to control, "last-resort" provisions are needed to reduce radioactive releases to amounts compatible with application of the Off-site Emergency Response Plan (i.e. a source term lower than S3):

- U2: location and correction of abnormal containment leak- tightness defects

and re-injection of highly radioactive liquid effluents back into the reactor building,
- U4: reducing releases into the environment to a minimum in the event of attack of the reactor building basemat by corium,
- U5: depressurization and filtration of the internal atmosphere of the containment to avoid its failure by slow pressure build- up,
- PAR: Passive Autocatalytic hydrogen Recombiners to reduce the consequences of a possible deflagration.

- U2 provisions

The U2 provisions provide for surveillance and, if necessary, restoration of the containment or a safeguard system liable to carry heavily contaminated water outside the containment after an accident. It uses the existing resources (surveillance is carried out using radiation sensors or water level sensors in the sumps).

The actions planned, which are performed at a very early stage, consists of:
- effecting or confirming isolation of the penetrations,
- location and restoration of leaktightness of any faulty penetrations,
- eventual re-injection into the reactor building of strongly radioactive liquid effluents (storage and processing using normal effluent processing resources could result in severe constraints concerning radiological protection of staff, evacuation conditions or the transport of radioactive waste) resulting from the containment. Leakage. Automatic triggering devices avoid a transfer of such effluents to the processing systems so they can be re-injected into the reactor building.

- U4 provisions

The corium could escape from the reactor vessel and attack the basemat. Design arrangements are therefore made to prevent direct contact between the corium and the environment, which could result in a source term outside the containment greater than 53. This consists in eliminating the drain system or basemat penetrations with liners under the reactor cavity. Direct contact between corium and the environment would therefore be delayed by at least a number of days (the time it is estimated that the corium would take to totally penetrate the basemat) which, associated with natural filtration by the soil, would result in a source term compatible with implementation of the Off-site Emergency Response Plan.
In the longer term, the time taken to reach the ground water would be sufficiently long to take any necessary restrictive measures concerning the use of water.

- U5provisions

Slow pressure build-up in the containment (resulting from vaporization of water in the sumps and possibly formation of incondensable gases created by decomposition of the basemat concrete by corium), when the design pressure was exceeded could result in delayed loss of its integrity.
The time available before loss of the confinement when its strength limits are reached varies between one and a number of days, depending on the hypotheses adopted. This proces5 leaves the operator sufficient time to take action to avoid failure of the containment while doing what it can to minimize radioactive releases. The action taken consists in depressurization with filtration, which is manually started before the internal pressure reaches its design basis pressure. This involves discharging part of the containment atmosphere through a metal screen in the

containment, then, after pressure reduction, filtering in a sand bed outside the containment, thus ensuring that radioactive releases will be well below the S3 source term.
The corresponding containment penetration is equipped with two valves operable from outside at a pressure greater than the containment design basis pressure. Gases are discharged via an independent line located in the gaseous radioactive effluent release stack.

- Hydrogen recombiners
Assessment of the behavior of the containment in this case is based on the deflagration of an arbitrary quantity of hydrogen corresponding to the oxidation of 75% of the active part of the fuel cladding.
In view of the strength limits of the containment and its isolation devices (there are substantial margins concerning its failure: the containment is guaranteed for the deflagration of a quantity of hydrogen produced by nearly 100% oxidation of the active part of the cladding), means of mitigation are not essential. However, as part of defense in depth, catalytic recombiners are to be installed, provided that it is demonstrated that they are effective and harm- legs.

B - RESIDUAL RISK
In view of the measures taken, both as concerns equipment and control, severe accidents involving short-term releases greater than the S3 source term form part of the residual risk. Generally speaking, for all such phenomena, R&D action as well as a watch on technological developments conducted by EDF, in association with the IRSN (technical support of French Nuclear Safety Authority) and the constructor in particular, ensure that this situation continues to apply and corresponds to international consensus.
The following situations thus constitute part of the residual risk. :
- High-pressure coremelt
The sequences liable to result in meltdown of the core with the primary system at high pressure are those resulting from a reactivity accident (causing an explosion of the fuel cladding), or inadequate core cooling without depressurization over a long period. The physical phenomena and the amounts of energy involved are different in the two cases, but the sequences all represent a threat to the containment in a relatively short period of time.
As concerns the risk of explosion of the fuel cladding, a number of sequences have been revealed by probabilistic safety assessments of the 900 and 1300 MWe series and by post-Chernobyl studies. These notably include the risk of massive heterogeneous dilution by the introduction of a slug of unborated water, and which may be cold, into the reactor core. This pure water could, for instance, come from the charging system: when the reactor coolant pumps are shut clown, it could accumulate in the primary loop and be propelled into the core when the pump restarts. Preventive measures (modification of equipment and operating procedures) have been taken to relegate this sequence to the residual risk.
Regarding the risks of slow meltdown with the primary system at pressure, rupture of the reactor vessel at high pressure would result in a substantial over-pressure in the reactor cavity and extremely high forces being exerted on the vessel supports, which could give rise to high-energy missiles. Furthermore, for some of these sequences, the conditions could be such as to cause a sudden dispersal of part of

the molten core in fine particles, liable to result in a sudden exchange of heat with the containment atmosphere and hence cause a large pressure peak (this phenomenon is referred to as Direct Containment Heating or DCH). Probabilistic safety assessments show that such events constitute parts of the residual risk, in view of the operating procedures used to reduce the pressure of the primary system to less than around 2 MPa (at which it is recognized there is no longer any risk).

- Hydrogen detonation

The main sources of hydrogen in a severe accident situation are:

- in the reactor vessel, oxidation of the zirconium alloy fuel cladding,
- outside the reactor vessel, oxidation of residual zirconium and other metals by steam, particularly in the event of attack of the basemat by the corium,
- in the longer term, radiolysis of water.

The risk engendered is loss of the containment in the event of generalised detonation if a sufficient quantity of hydrogen is formed or, even, a deflagration to detonation transition (the risk of deflagration is taken into account, see §2.4).
The problem is, in fact, ground for concern essentially for containments which have a small free internal volume (ice condenser containments for instance), or those designed for low pressure peaks (pressure suppression containments of boiling water reactors), but is far less so for large, dry containments designed to withstand high pressures, such as the French containments. For the latter, international consensus is that the risk of generalized detonation is sufficiently low to be disregarded. As concerns deflagration to detonation transition, the low "bunkering" of French containments makes the probability negligible.

- Steam explosion

The containment could be ruptured in the event of violent interaction between molten fuel and the water contained in the lower part of the reactor vessel, or in the reactor cavity if the bottom of the vessel should fail. International studies have shown that, even with a conservative assessment of the amount of energy released in the event of such interaction within the pressure vessel, a pres- sure peak resulting from these phenomena or the release of massive missiles liable to jeopardize the integrity of the containment are implausible. This being the case, it can be stated that this mode of failure of the containment constitutes part of the residual risk.

- By-passing or non-isolation of the containment

The situations involved essentially consist of steam generator tube ruptures and substantial leaks in the safety injection system and containment spray system recirculation loops with the core in a degraded condition.
In view of the operating strategy adopted (keeping the secondary side of the steam generators primed with water), the thermal stresses on the steam generator tubes remain smal1. Dry out of the steam generators in such a situation constitutes part of the residual risk.
As concerns the risk of leakage from the safety injection system and containment spray system recirculation loops, it has been verified that the equipment concerned will remain leaktight up to pressures and temperatures representative of severe accident situations.

Concerning a severe accidents reference system, there is at the moment no clearly defined objectives being the subject of a consensus between EDF and the Nuclear Safety Authority. The « severe accident » reference system was reviewed for the fist time during the meeting of the Advisory Committee held on March 31, 2005. It is clear that all is based on the releases in the environment, which must remain compatible with the off-site emergency plan. Thus, EDF has proposed probabilistic and radiological objectives in its reference system for severe accidents. They will be examined at the time of the next Advisory Committee meeting on the radiological consequences and the severe accidents, which is scheduled in 2006. One has to note that within the preceding Advisory Committee framework relating to the severe accidents, ASN required that EDF set up various equipment of prevention and mitigation of these risks such as the installation of hydrogen « recombiners ».

| Seq. No 197 | Country | Article Article 19.4 | Ref. in National Report |
|---|---|---|---|

Question/ Comment: Are the instrumentation and control systems (I&C) for safety relevant functions at NPPs certified for harsh environmental conditions occurred during potential severe accidents? If yes, what is the specification of a harsh environmental condition?

Answer: The instrumentation and control systems (I&C) needed for the mitigation of severe accidents are identified. The behaviour of those components submitted to harsh environmental conditions in these types of accidents is checked by representative tests.
For each relevant component, a specification of a harsh environmental condition is established, based mainly on the time the equipment is needed in severe accident and its geographical position into the containment, which determines the dose rate and the integrated dose.

| Seq. No 198 | Country | Article Article 19.4 | Ref. in National Report page 130 |
|---|---|---|---|

Question/ Comment: In 2002, EDF forwarded a draft "severe accident" reference system to the ASN. Please provide more detailed information about the objective and specific features of the severe accident reference system.

Answer: The purpose of the "severe accident" reference system is to define a coherent set of requirements making it possible to guarantee a good level of safety of the French NPP units with respect to the severe accidents. It is made up of:

- objectives known as "high level" aiming at guaranteeing a sufficient level of safety. They consist, on the one hand, in probabilistic objectives and, on the other hand, in objectives of radiological consequences, aiming at showing either the hypothetical character, or the acceptable character with respect to the population of the scenarios of severe accident. In addition objectives of protection against radiation of the personnel are defined to evaluate the "operable" character of the sections after a severe accident.
- description of the equipments, operating procedures or guidelines necessary to mitigate the severe accidents and requirements (or objective determinists) on these materials (temperature and pressure of operation, doses taken by the materials, containment integrity,....). These equipments and the associated

requirements are defined on the basis of identification of the dominating physical phenomena and the evaluation of the course of the accidents (construction of the scenarios) and of the inherent risks (probability of the scenarios and forecast of the radioactive releases).

A Level 2 PSA is then used to check the respect of the probabilistic and radiological consequences objectives.

The « severe accident » reference system was reviewed for the fist time during the meeting of the Advisory Committee held on March 31, 2005.The reference system constitutes first of all an inventory of subjects having been examined within the preceding Advisory Committee framework relating to the severe accidents (before that of 2004). It presents a synthesis of the studies relating to severe accidents with assumptions and conclusions which lead to requests for engineering changes or evolutions of accidental or post-accidental procedures.
In this reference system, is also indicated the approach and the aims followed by EDF for severe accidents prevention and mitigation.
After discussions with EDF on this reference system which is still at the moment at the elaboration state, ASN wishes that this document in the long term becomes a reference frame of requirements relating to the severe accidents applicable and opposable to the sites, making it possible to check the conformity of 900 MWe reactors after their third 10-yearly outage.

| Seq. No 199 | Country | Article Article 19.6 | Ref. in National Report 19.2.6 |
|---|---|---|---|

Question/ Comment

Reference :19.2.6 Declaration of anomalies and incidents by EDF
"a file is created and updated for each significant anomaly and incident and contains, amongst other things, the results of this analysis and EDF keeps the ASN regularly informed of the state of this file"

Q1: Would you explain a little bit more about the EDF's information reporting system to the ASN?

Answer

Each NPP sets up for every declared significant event an analysis report which is released to the Nuclear Safety Authority within two months after the declaration. If complementary analysis elements occur at a later stage the report is updated. A computer data base (SAPHIR) gathers together all the historical data of the events and the interventions, the Nuclear Safety Authority has access to this data base for events that are significant for safety, the environment and for radioprotection, as well events concerning safety but that are of lesser importance.

As a complement, every quarter, a meeting is organized between EDF and ASN to present the Experience Feed Back (REX) and its acknowledgement by EDF, on the basis of the events entered into SAPHIR. A report of the meeting is drawn up in which EDF presents the processing carried out for the salient events and answers the questions put by ASN regarding the events it has selected for its own analysis

| Seq. No 200 | Country | Article Article 19.7 | Ref. in National Report 19.2.7, p122 |
|---|---|---|---|

Question/ It is recognized that EDF is a large organization with a high volume of internal

Comment operating experience to share between its operating reactor units.
What criteria does EDF use to screen the relevance of operating experience arising outside EDF (outside France)? How many significant events arising from operating experience outside France have been included in the information shared by EDF with its individual plants?

Answer The criteria that EDF uses for analyzing an external event are:

- Event marking an international level (classification 2+ on the INES scale, for instance or having caused major media spinoff); e.g.: damage to 30 fuel rod assemblies in the Paks plant in April 2003, fatal accident in Mihama in 2004.
- Discovery of faults in equipment important for safety in particular the reactor and its auxiliaries; for instance: leak in vessel bottom head in South Texas in 2003.
- Events of interest for EDF for which the degree of reproducibility has to be evaluated; e.g: event provoked by common mode fault,
- The Significant Event Report and Significant Operating Experience Report released by WANO

The number of foreign events selected by corporate level and disseminated toward the NPP sites comes to around 10 and 15 by year, other than SER/SOER WANO. The EDF sites are also all provided with access to the Internet site at WANO and can consult the event data base.

The feedback from experience with international events is also generalized in the Nuclear Coordination Division (NPD) analyses (similar past event research).

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 201 | | Article 19.7 | P122.Ch19.2.7 |

Question/ Comment How to deal with recurring significant events by the safety authority?

Answer The Significant Events declared by the NPP sites are all the subject of local analysis which consists more particularly in identifying the deep-lying causes and in defining the corrective actions that may be required at site level to avoid its recurrence. Some of them give rise to complementary development regarding the inventory, depending on the importance they represent regarding experience feedback.
The events presented in the annual safety balance which is sent to the Nuclear Safety Authority are retained as a function of:

- their intrinsic safety such as events that can have unacceptable consequences from the safety standpoint (risk of damage to the core or dumping into the environment), whether they initiate incidental or accidental transient effects or disclose defective states in the quality equipment or system,
- their belonging to repetitive deviation families.

The selected events are all classified according to three criteria :

- the marking events in terms of importance (the method for selecting and evaluating the marking events by quantity is based on the PSA),
- the marking events by safety function (reactivity, cooling, containment),
- the repetitive deviation families by characteristic theme of a defective quality

system state (e.g.: nonconformity with technical operating specifications or periodic tests, alignment or shutting down faults, etc).

For its part the ASN organises every three years a dedicated session on that topic of the Advisory Committee of experts for nuclear reactors.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 202 | | Article 19.7 | 19.2.7 P122 |

Question/ Comment

Reference :19.2.7 Operating feedback at EDF
In EDF, significant events that have the greatest effect on safety (around 40 a year) require assessment of the potential risk of core damage,

Please give us supplementary explanation about the contents of the 40 significant events.
Q1: What kind of state is applicable " the greatest effect on safety "? Please let us know the definition of "the greatest effect on safety".
Q2: Please let us show some examples of the significant events recorded around 40 events a year.
Q3: Please let us know about the organization, the methods and software tools for probabilistic analysis with regard to core damage applied to the 40 significant events.
Q4: Regarding the probabilistic analysis with regard to core damage, how long does it take to complete the analysis? Is the analysis conducted in each plant?

Answer

Q1/
The significant events having "the highest impact on safety" (which are named " salient events") are selected by EDF using deterministic criteria among the Significant Events factors or any other event or feedback considered to be worthwhile. These criteria, of which there are 10, can be classified into four different sections.

- Incidental transients whose development gave rise to a problem or accidental transients.
- Significant effects concerning the safety-related lines of defence (material or human).
- Generic events having a real or potential safety impact.
- Judgement of an expert from a probabilistic point of view.

Q2/
Here are a few examples of events appearing in the list of salient events that appear from the year 2003:

- Presence of foreign loose parts in a cold branch injection line.
- Triggering-off of a emergency diesel generator due to a fuel hose failure.
- Deviation in adjustment of electrical protection on a Containment Spray System motor pump.
- Inadvertent safety injections during Shutdown on Residual Heat Removal System.
- Component Cooling System valve left in wrong position due to an alignment error.

Q3/
The probabilistic analysis of events to be contemplated consists in envisioning from the observed situation the potential scenarios of degradation that could lead to core damage. This analysis is applied to certain significant and Safety related Events (EIS) probably selected on deterministic grounds. These are "salient" events, i.e. for which it might seem that the risk increase was significant or for which probabilistic analysis is likely to produce worthwhile quality information. The seriousness of the salient events of a suitable nature is quantified from a probabilistic point of view on the basis of data obtained from the Probabilistic Safety Assessment (PSA). The value obtained is referred to as the Potential Risk Index (IRP) and represents the conditional probability of core damage while bearing in mind that the event took place. The events for which the potential risk index is in excess of 10-6 are referred to as precursor events (precursors to severe accidents, i.e. that could lead to core damage with a high probability level).
The identification of the precursors addresses three concerns:

- Knowing the situations where the risk of fuel damage was highest.
- Establishing priorities for corrective actions.
- Disseminating a probabilistic useful safety culture in other fields (risk analysis, decision-making in particular).

Q4/
The probabilistic analysis process of events referred to as the "Precursor Program" is broken down on an annual basis and the conclusions are presented at the beginning of the following year. The steps are as follows:

- Publication of event report by the unit at the origin,
- Examination by the EDF corporate division in charge of the event-related feedback,
- Selection of salient events,
- Complements of technical analysis,
- Probabilistic analysis (support for the RISK SPECTRUM computer application),
- Establishing of the balance sheet.

The production of probabilistic data is the responsibility of EDF corporate engineering units and the NPP units use them under corporate control.

| Seq. No 203 | Country | Article Article 19.7 | Ref. in National Report 19.4.1 P136 |
|---|---|---|---|

Question/ Comment: Reference : 19.4.1 power reactor operation

Q1: Please let us know about the regulatory organization, the methods and software tools for risk analysis.

Answer: The risk can be defined by the couple probability versus consequence. ASN considers that deterministic and probabilistic approaches are both relevant and complementary to address the risk. Although PSAs are a useful tool in order to quantitatively measure the risk, there are some aspects such as the safety culture that cannot be measured in a quantitative manner. For instance, the Dampierre NPP was put under a reinforced surveillance in 2002 on the basis of a qualitative judgement made by ASN, not on the basis of PSA.

| Seq. No | Country | Article | Ref. in National Report |
|---|---|---|---|
| 204 | | Article 19.7 | 19.4.1.3.2,P129 |

Question/ Comment

Reference : 19.4.1.3.2 Reactor operation in severe accident situations
In 2002, EDF forwarded a draft "severe accident" reference system to the ASN. This draft reference system has been submitted to the Advisory Committee for nuclear reactor for examination in 2004.

Q1: Please explain the "Severe Accident Reference System" specifically. In what manner is the system used by the regulator?
Q2: How many events, in the 100 of INES level-1 events from 1998 to 2000, were caused by human factor or by degradation of safety culture? Please explain outline and cause of these events.
Q3: Please let us know the contents of the approach and target of risk prevention and risk mitigation for major accidents?

Answer

Q1 – Read answer to question No 198:
The purpose of the "severe accident" reference system is to define a coherent set of requirements making it possible to guarantee a good level of safety of the French NPP units with respect to the severe accidents. It is made up of:

- objectives known as "high level" aiming at guaranteeing a sufficient level of safety. They consist, on the one hand, in probabilistic objectives and, on the other hand, in objectives of radiological consequences, aiming at showing either the hypothetical character, or the acceptable character with respect to the population of the scenarios of severe accident. In addition objectives of protection against radiation of the personnel are defined to evaluate the "operable" character of the sections after a severe accident.
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A Level 2 PSA is then used to check the respect of the probabilistic and radiological consequences objectives.

This severe accident reference system is still under discussion with Nuclear Safety Authority . The Regulator does not use it.

Concerning severe accidents, there is at the moment no clearly defined objectives being the subject of a consensus. It is clear that all is based on the releases in the environment which must remain compatible with the off-site emergency plan. Thus, EDF has proposed probabilistic and radiological objectives in its reference system for severe accidents. They will be examined at the time of the next

Advisory Committee meeting on the radiological consequences and the severe accidents, which is scheduled in 2006. One has to note that within the preceding Advisory Committee framework relating to the severe accidents, ASN required that EDF set up various equipment of prevention and mitigation of these risks such as the installation of hydrogen « recombiners ».

The « severe accident » reference system was reviewed for the fist time during the meeting of the Advisory Committee held on March 31, 2005.The reference system constitutes first of all an inventory of subjects having been examined within the preceding Advisory Committee framework relating to the severe accidents (before that of 2004). It presents a synthesis of the studies relating to severe accidents with assumptions and conclusions, which lead to requests for engineering changes or evolutions of accidental or post-accidental procedures.
In this reference system, is also indicated the approach and the aims followed by EDF for severe accidents prevention and mitigation.
After discussions with EDF on this reference system which is still at the moment at the elaboration state, ASN wishes that this document in the long term becomes a reference frame of requirements relating to the severe accidents applicable and opposable to the sites, making it possible to check the conformity of 900 MWe reactors after their third 10-yearly outage.

Q2/
Ratio or INES level-1 events from 1998 to 2000, caused by human factor (or by degradation of safety culture) are shown in this table (number, %):

| Year | 1998 | | 1999 | | 2000 | | ! | 1998-2000 | |
|---|---|---|---|---|---|---|---|---|---|
| Human factor | 47 | 75 % | 46 | 67 % | 69 | 72 % | ! | 162 | 71 % |
| Component failures | 14 | 22 % | 20 | 29 % | 24 | 25 % | ! ! | 58 | 25 % |
| Non safety related | 2 | 3 % | 3 | 4 % | 3 | 3 % | ! ! | 8 | 4 % |
| Total | 63 | | 69 | | 96 | | ! | 228 | |

The main root causes are :
- Non compliance with Operating Technical Specifications,
- Alignment error,
- Non compliance with derogatory safety regulation,
- Non compliance with test periodicity,
- Errors of parameters implementation.

Q3/ Read answer to question No 196/2:

MITIGATION: Allowance for certain severe accidents

The safety approach is supplemented by allowance for a nuclear steam supply system state characterized by core degradation, with the primary system depressurized (operating conditions designated "non-plausible"). The principle of the approach is to have available the necessary resources to delay and limit the consequences of events leading to such a state (maintaining, adequate confinement for a sufficiently long period), to enable *the* public authorities to take off-site protective action (application of the Off-site Emergency Response Plan which

notably. includes the evacuation of people within a 5 km radius and sheltering within a 10 km radius, within 12 to 24 hours).
Introducing guidelines and procedures calling for available conventional resources and special “last- resort” resources carries this out:

- U4 provisions which avoid early contact between corium and the environment if the former attacks the reactor building basemat,
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It is to be noted that allowance for severe accidents closely links the "design" and "operation" aspects.

A- SEVERE ACCIDENT PROVISIONS

The S1 source term (representative of accidents with early loss of containment) corresponds to high-energy accidents that are of extremely low probability and therefore for which it is not necessary to make special provisions.

As the potential consequences of the S2 source term (representative of accidents involving release from the containment directly into the atmosphere after delayed loss of leaktightness: at least one day after the accident) can be difficult to control, "last-resort" provisions are needed to reduce radioactive releases to amounts compatible with application of the Off-site Emergency Response Plan (i.e. a source term lower than S3):

- U2: location and correction of abnormal containment leak- tightness defects

and re-injection of highly radioactive liquid effluents back into the reactor building,
- U4: reducing releases into the environment to a minimum in the event of attack of the reactor building basemat by corium,
- U5: depressurization and filtration of the internal atmosphere of the containment to avoid its failure by slow pressure build- up,
- PAR: Passive Autocatalytic hydrogen Recombiners to reduce the consequences of a possible deflagration.

- U2 provisions

The U2 provisions provide for surveillance and, if necessary, restoration of the containment or a safeguard system liable to carry heavily contaminated water outside the containment after an accident. It uses the existing resources (surveillance is carried out using radiation sensors or water level sensors in the sumps).

The actions planned, which are performed at a very early stage, consists of:
- effecting or confirming isolation of the penetrations,
- location and restoration of leaktightness of any faulty penetrations,
- eventual re-injection into the reactor building of strongly radioactive liquid effluents (storage and processing using normal effluent processing resources could result in severe constraints concerning radiological protection of staff, evacuation conditions or the transport of radioactive waste) resulting from the containment. Leakage. Automatic triggering devices avoid a transfer of such effluents to the processing systems so they can be re-injected into the reactor building.

- U4 provisions

The corium could escape from the reactor vessel and attack the basemat. Design arrangements are therefore made to prevent direct contact between the corium and the environment, which could result in a source term outside the containment greater than 53. This consists in eliminating the drain system or basemat penetrations with liners under the reactor cavity. Direct contact between corium and the environment would therefore be delayed by at least a number of days (the time it is estimated that the corium would take to totally penetrate the basemat) which, associated with natural filtration by the soil, would result in a source term compatible with implementation of the Off-site Emergency Response Plan.
In the longer term, the time taken to reach the ground water would be sufficiently long to take any necessary restrictive measures concerning the use of water.

- U5provisions

Slow pressure build-up in the containment (resulting from vaporization of water in the sumps and possibly formation of incondensable gases created by decomposition of the basemat concrete by corium), when the design pressure was exceeded could result in delayed loss of its integrity.
The time available before loss of the confinement when its strength limits are reached varies between one and a number of days, depending on the hypotheses adopted. This proces5 leaves the operator sufficient time to take action to avoid failure of the containment while doing what it can to minimize radioactive releases. The action taken consists in depressurization with filtration, which is manually started before the internal pressure reaches its design basis pressure. This involves discharging part of the containment atmosphere through a metal screen in the

containment, then, after pressure reduction, filtering in a sand bed outside the containment, thus ensuring that radioactive releases will be well below the S3 source term.
The corresponding containment penetration is equipped with two valves operable from outside at a pressure greater than the containment design basis pressure. Gases are discharged via an independent line located in the gaseous radioactive effluent release stack.

- Hydrogen recombiners
Assessment of the behavior of the containment in this case is based on the deflagration of an arbitrary quantity of hydrogen corresponding to the oxidation of 75% of the active part of the fuel cladding.
In view of the strength limits of the containment and its isolation devices (there are substantial margins concerning its failure: the containment is guaranteed for the deflagration of a quantity of hydrogen produced by nearly 100% oxidation of the active part of the cladding), means of mitigation are not essential. However, as part of defense in depth, catalytic recombiners are to be installed, provided that it is demonstrated that they are effective and harm- legs.

B - RESIDUAL RISK
In view of the measures taken, both as concerns equipment and control, severe accidents involving short-term releases greater than the S3 source term form part of the residual risk. Generally speaking, for all such phenomena, R&D action as well as a watch on technological developments conducted by EDF, in association with the IRSN (technical support of French Nuclear Safety Authority) and the constructor in particular, ensure that this situation continues to apply and corresponds to international consensus.
The following situations thus constitute part of the residual risk. :
- High-pressure coremelt
The sequences liable to result in meltdown of the core with the primary system at high pressure are those resulting from a reactivity accident (causing an explosion of the fuel cladding), or inadequate core cooling without depressurization over a long period. The physical phenomena and the amounts of energy involved are different in the two cases, but the sequences all represent a threat to the containment in a relatively short period of time.
As concerns the risk of explosion of the fuel cladding, a number of sequences have been revealed by probabilistic safety assessments of the 900 and 1300 MWe series and by post-Chernobyl studies. These notably include the risk of massive heterogeneous dilution by the introduction of a slug of unborated water, and which may be cold, into the reactor core. This pure water could, for instance, come from the charging system: when the reactor coolant pumps are shut clown, it could accumulate in the primary loop and be propelled into the core when the pump restarts. Preventive measures (modification of equipment and operating procedures) have been taken to relegate this sequence to the residual risk.
Regarding the risks of slow meltdown with the primary system at pressure, rupture of the reactor vessel at high pressure would result in a substantial over-pressure in the reactor cavity and extremely high forces being exerted on the vessel supports, which could give rise to high-energy missiles. Furthermore, for some of these sequences, the conditions could be such as to cause a sudden dispersal of part of

the molten core in fine particles, liable to result in a sudden exchange of heat with the containment atmosphere and hence cause a large pressure peak (this phenomenon is referred to as Direct Containment Heating or DCH). Probabilistic safety assessments show that such events constitute parts of the residual risk, in view of the operating procedures used to reduce the pressure of the primary system to less than around 2 MPa (at which it is recognized there is no longer any risk).

- Hydrogen detonation

The main sources of hydrogen in a severe accident situation are:

- in the reactor vessel, oxidation of the zirconium alloy fuel cladding,
- outside the reactor vessel, oxidation of residual zirconium and other metals by steam, particularly in the event of attack of the basemat by the corium,
- in the longer term, radiolysis of water.

The risk engendered is loss of the containment in the event of gene- **1**ralised detonation if a sufficient quantity of hydrogen is formed or, even, a deflagration to detonation transition (the risk of deflagration is taken into account, see §2.4).
The problem is, in fact, ground for concern essentially for containments which have a small free internal volume (ice condenser containments for instance), or those designed for low pressure peaks (pressure suppression containments of boiling water reactors), but is far less so for large, dry containments designed to withstand high pressures, such as the French containments. For the latter, international consensus is that the risk of generalized detonation is sufficiently low to be disregarded. As concerns deflagration to detonation transition, the low "bunkering" of French containments makes the probability negligible.

- Steam explosion

The containment could be ruptured in the event of violent interaction between molten fuel and the water contained in the lower part of the reactor vessel, or in the reactor cavity if the bottom of the vessel should fail. International studies have shown that, even with a conservative assessment of the amount of energy released in the event of such interaction within the pressure vessel, a pres- sure peak resulting from these phenomena or the release of massive missiles liable to jeopardize the integrity of the containment are implausible. This being the case, it can be stated that this mode of failure of the containment constitutes part of the residual risk.

- By-passing or non-isolation of the containment

The situations involved essentially consist of steam generator tube ruptures and substantial leaks in the safety injection system and containment spray system recirculation loops with the core in a degraded condition.
In view of the operating strategy adopted (keeping the secondary side of the steam generators primed with water), the thermal stresses on the steam generator tubes remain smal1. Dry out of the steam generators in such a situation constitutes part of the residual risk.
As concerns the risk of leakage from the safety injection system and containment spray system recirculation loops, it has been verified that the equipment concerned will remain leaktight up to pressures and temperatures representative of severe accident situations.

Concerning a severe accidents reference system, there is at the moment no clearly defined objectives being the subject of a consensus between EDF and the Nuclear Safety Authority. The « severe accident » reference system was reviewed for the fist time during the meeting of the Advisory Committee held on March 31, 2005. It is clear that all is based on the releases in the environment, which must remain compatible with the off-site emergency plan. Thus, EDF has proposed probabilistic and radiological objectives in its reference system for severe accidents. They will be examined at the time of the next Advisory Committee meeting on the radiological consequences and the severe accidents, which is scheduled in 2006. One has to note that within the preceding Advisory Committee framework relating to the severe accidents, ASN required that EDF set up various equipment of prevention and mitigation of these risks such as the installation of hydrogen « recombiners ».

| Seq. No 205 | Country | Article Article 19.7 | Ref. in National Report |
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Question/ Comment: Is there interrelation or impact of the reactor operating experience on the current energy market?

Answer: Globally, approximately 80% of the electric energy in France is of nuclear origin and is produced by EDF power plants. EDF has therefore developed load follow-up for most of its plant units. As a complement, the plant shutdown planning is controlled at the overall level of the Company and the plant unit production program is optimized according to all the data and the demands. The Nuclear Production Division also has to supply services to the network manager, for instance, for frequency and voltage adjustment, and for robustness with respect to network incidents. As a complement, EDF is particularly vigilant regarding the control of fortuitous unavailability by preventive actions. The opening of the electricity market has no fundamental effect on the operating context but the associated economic challenges may be greater.

| Seq. No 206 | Country | Article Article 19.8 | Ref. in National Report 19.1.3, 1 para |
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Question/ Comment: Would you list the main activities (apart from those specified by ASN) to be performed prior to Unit decommissioning?

Answer: Between Operation and dismantling there is a phase of "post operation", which begins with the shutdown of the installation, comprises simplification of the facilities, before definitive shutdown of the equipment. The simplification results in an adaptation of the equipment to definitive shutdown state, which requires an analysis of the functional requirements of the elementary system, before analysis and execution of modifications to adapt to the new requirements. Then the installations whose functions are no longer required pass from their operational configuration to a ready-for-dismantling state (Definitive shutdown of equipment). Once the equipment is definitively shutdown and made safe for this operation (electrical isolation, chemical cleaning …), dismantling of complete zones or buildings can began.
Definitive shutdown of equipment is an irreversible act : the system will no longer be able to operate ; it is separated from the rest of the NPP. It implies the following actions :

1. Shutdown
2. Fluids removed and circuits cleaned
3. Mechanical isolation
4. Electrical isolation
5. Elimination of risks (chemical, radiological…)
6. Elimination of I & C
7. Treatment of documentation

Once definitively shutdown, a functional system does no longer require any monitoring or conservation measures. The residual risks related to dismantle are clearly identified. A process of local marking is used: the equipment to be definitively shutdown is identified by indication « MHSD » (definitive shutdown – in French) and the file number.

| Seq. No 207 | Country | Article Article 19.8 | Ref. in National Report page 111f |
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| Question/ Comment | What is the concept for treatment of spent fuel? If it is only reprocessing, reprocessing waste should be regarded as (indirect) operational waste and be addressed in the report. |
| Answer | The waste to be addressed in this report is "waste directly related to the operation and on the same site as that of the nuclear installation" (Art. 19 viii). Since spent fuel is not reprocessed on the same site, the resulting waste does not fall under this Convention.<br>However these waste fall under the scope of the Joint Convention where its management is presented and discussed exhaustively. Therefore, for an exhaustive view on the safety of spent fuel and radioactive management in France outside NPPs, please refer to the 1st France Report for the Joint Convention (May 2003) and further to its 2nd report (October 2005). |

**Planned activities to improve safety**

| Seq. No 208 | Country | Article Planned Activities | Ref. in National Report |
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| Question/ Comment | The report states that the Nuclear Safety Authority prioritise the ensuring of better considerations of environmental issues. Are there particular environmental issues requiring better consideration? Which measures for ensuring better consideration of environmental issues have been developed, implemented and applied already? | | |
| Answer | The former regulation had time limited licenses for water intake and effluent release permits therefore all these licenses had or will have to be renewed from time to time. A review of most of the environmental licenses has also been planed.<br><br>For each concerned site an impact study has been required. Theses impact studies contributed to build new licenses with lower limits for liquid and gaseous radioactive effluents. A particular action is led on decreasing chemical releases limits. In order to prepare these licenses technical appraised is required.<br><br>Indeed, as it was discussed for a long time after the Country Group 2 rapporteur's presentation during the Plenary session of the second CNS review meeting, there are good reasons to revise effluent release limits to lower values.<br><br>Firstly it is now a common practice for chemical release to try to set the limits as low as reasonably achievable, based on the precaution principle.<br><br>Then, in the same way it was found that the older limits for radioactive releases were originally based only on human health detrimental effect without any special consideration as regards the environment and remained much higher than needed.<br><br>The application of the same precaution principle led the ASN to require that new effluent limits be as low as reasonably needed for the operation of nuclear installations. | | |