ASN/DCN/ Report No. 0080-2007
presented in its final form to the ASN Board, on 16 February 2007

Nuclear Power Plant
Division

Nuclear Pressure Equipment
Division

# Authorisation Decree Application for the creation of the FLAMANVILLE-3 Basic Nuclear Installation

## Executive Summary of the Technical Review

| | **DCN Line management approval(s)** | |
|---|---|---|
| Written by | Assistant to the Director | Director |
| Ph. Dupuy | P. Charpentier | O. Gupta |
| | | |

| **Line management approval(s)** | |
|---|---|
| Deputy Director-General | Director-General |
| A. Schmitt | J.-C. Niel |
| | |

| | **DEP Line management approval(s)** | |
|---|---|---|
| Written by | Assistant to the Director | Director |
| F. Colonna<br>S. Gitkoff<br>P. Mutin | J.-C. Van Hoecke | S. Mourlon<br>P.i S. Limousin |
| | | |

## Table of contents

**Purpose of the report**

On 9 May 2006, *Électricité de France* (EDF) submitted to the Ministers for Nuclear Safety an authorisation decree application for an EPR-type reactor on the site of the Flamanville Nuclear Power Plant (NPP).

Article 29 of *Act No. 2006-686 of 13 June 2006 on Transparency and Security in the Nuclear Field* prescribes that the creation of any basic nuclear installation shall be issued by a decree taken after consultation with the Nuclear Safety Authority (*Autorité de sûreté nucléaire* – ASN).

The purpose of this report is to provide ASN's Board with a summary of the technical review led by ASN services and carried out by their technical support agencies, namely the IRSN[1], the GPR[2] and the Standing Nuclear Section of the CCAP[3] between 2001 and 2006.

After summing up the conclusions of the review on the safety options of the European Pressurized Reactor (EPR) Project, as carried out between 1993 and 2000, this report describes the process and modalities of the review conducted from 2001 to 2006. Besides providing the opinion of ASN's services on the creation-licence application, it also outlines the further review to be carried out, if the authorisation decree is issued.

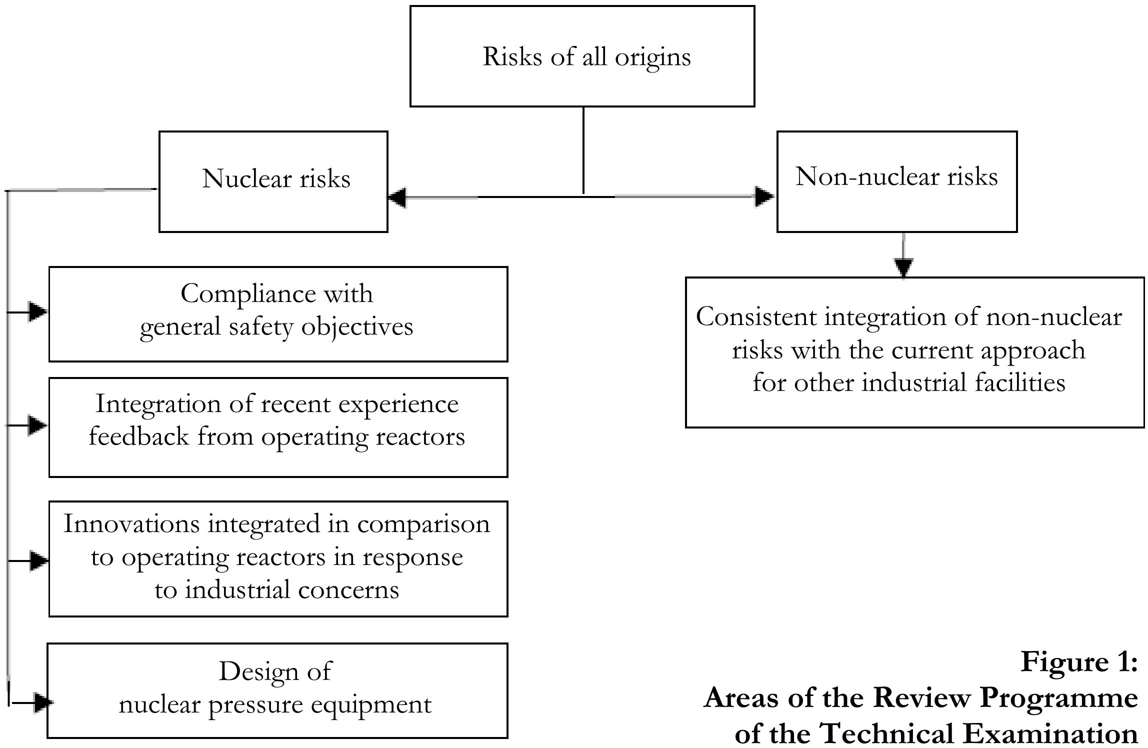



**Figure 1:**
**Areas of the Review Programme of the Technical Examination from 2001 to 2006**

---

1. *Institut de radioprotection et de sûreté nucléaire* (French Institute for Radiation Protection and Nuclear Safety).
2. *Groupe permanent d'experts pour les réacteurs nucléaires* (Advisory Expert Group on Nuclear Reactors).
3. *Commission centrale des appareils à pression* (Central Committee for Pressure Vessels).

## I. Conclusions of the safety-option review

Initiated after the German and French nuclear safety authorities determined safety objectives for the new generation of pressurised-water reactors[4] (PWR), in 1993, the review of the safety options of the EPR Project ended in October 2000 with the adoption by the GPR and associated German experts of a document entitled "Technical Guidelines for the Design and the Construction of the Next Generation of Nuclear Power Plants with Pressurized Water Reactors".

Those guidelines restate in a structured and organised fashion the overall technical recommendations made by French and German experts and validated by the ASN throughout the safety-option review. As such, they constituted the main element of the technical reference system for reviewing the EPR Project between 2001 and 2006.

The guidelines were formalised in a letter addressed to the President of EDF (Reference [1]) in which the French government considered that the reviewed safety options proved satisfactory in relation to the set objective to improve the overall safety of nuclear reactors in service.

The letter emphasised the need to confirm that assessment by the analysis by a certain number of detailed design studies of particular interest, in the framework of the review of a potential authorisation decree application for an EPR reactor in France, as follows:

(1) the prevention of human errors, the improvement in the radiation protection of workers, as well as the reduction of radioactive releases and of the quantity and activity of the waste involved;

(2) design, manufacturing and operating provisions for the main lines of the primary circuit and, if need be, of secondary circuits, with a view to excluding their full double-ended break from certain accident studies;

(3) the physical architecture of the instrumentation and control systems;

(4) the design of the core catcher set in place for managing severe accidents;

(5) the compatibility of the characteristics of the standard EPR Project with the proposed implementation site, and

(6) the protection of the installation against malevolent events.

From an overall standpoint, the letter stated lastly that changes to the project might be deemed necessary at the request of public authorities in case a significant evolution in safety requirements was justified in order to reflect new elements resulting especially from experience in design, construction or operation of existing reactors or, more generally, from advances in knowledge about safety.

4. Those safety objectives are described in Annex IV.

## II. Context of the Technical Review between 2001 and 2006

### *II.1 From 2001 to 2003*

Between 2001 and 2003, the technical review continued according to the modalities set in place since 1998 (Figure 2) with two meetings of the GPR[5] and the associated German experts (GPR meetings of 3 July 2002 and 3 July 2003.

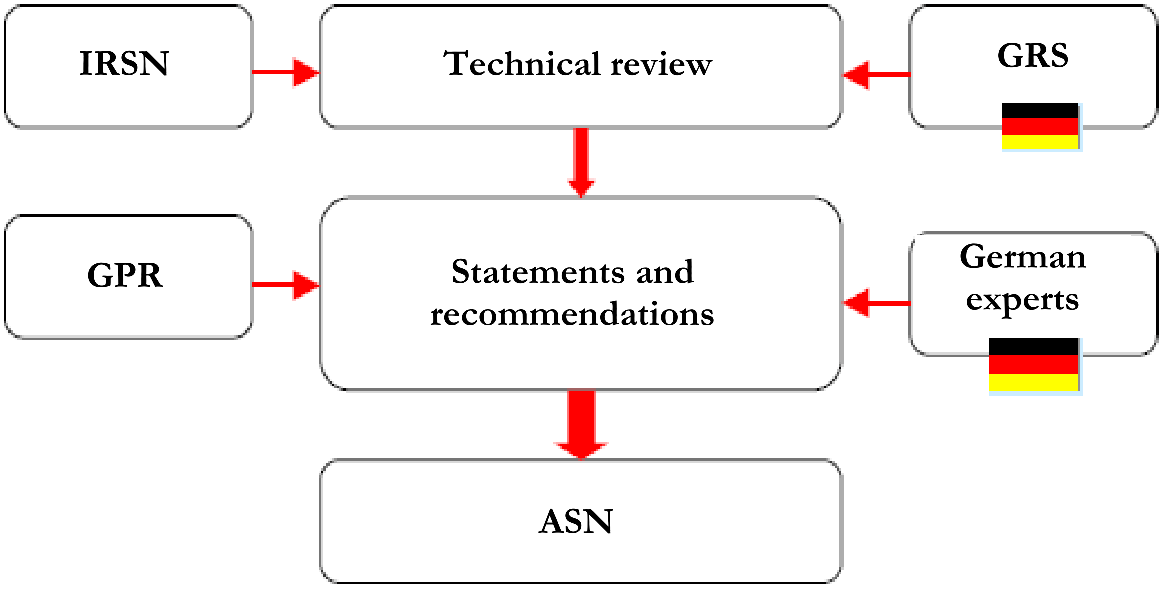



**Figure 2**

In parallel, the review of the choices for the design and manufacturing of nuclear pressure equipment started with the meeting of the Standing Nuclear Section (*Section permanente nucléaire* – SPN), held on 2 July 2003.

### *II.2 In 2004-05*

Years 2004 and 2005 were marked by a rapid evolution in the context of the EPR Project. In 2004, for instance, a contract was signed in Finland between the French company AREVA and the Finnish Company *Teollisuuden Voima Oy* (TVO) for the supply of an EPR-type reactor on the Olkiluoto Site, while French Parliament adopted the *Planning Act No. 2005-781 of 13 July 2005 Setting Forth the Orientations of the Energy Policy,* thus maintaining open the nuclear option until 2020.

Starting in 2004, the technical review of the EPR Project increased its pace and concentrated not only on analysing the study notes provided by EDF, but also, in preparation for a authorisation decree application, on the successive versions of the preliminary safety report (Version 1 of 28 January 2004; Version 2 of 7 October 2005 and Version 3 of 9 March 2006.

In 2004, the GPR and the associated German experts met twice at ASN's request on 1 July and 18 November, respectively.

In parallel to the technical review of the EPR Project, ASN also launched in 2004 a review of the methodology for assessing the radiological impact of accidents that EDF intends to use for operating reactors and EPRs.

5. The review process is presented in more detail in Annex II.

In 2005, the GPR and the associated German experts also met twice at ASN's request on 5 July and 1 December, respectively.

Such evolution in the context and such acceleration in the technical review of the Project lead to changes in the review partnerships in place (Figure 3), as follows:

- the IRSN is now the only entity to review the technical documents provided by EDF, and
- an expert from the Finnish Radiation and Nuclear Safety Authority (*Säteilyturvakeskus* –STUK), already involved in the review of the EPR Project at Olkiluoto 3, is appointed in 2004 as a member of the GPR and participates in the preparation of the opinions and recommendations to be formulated.

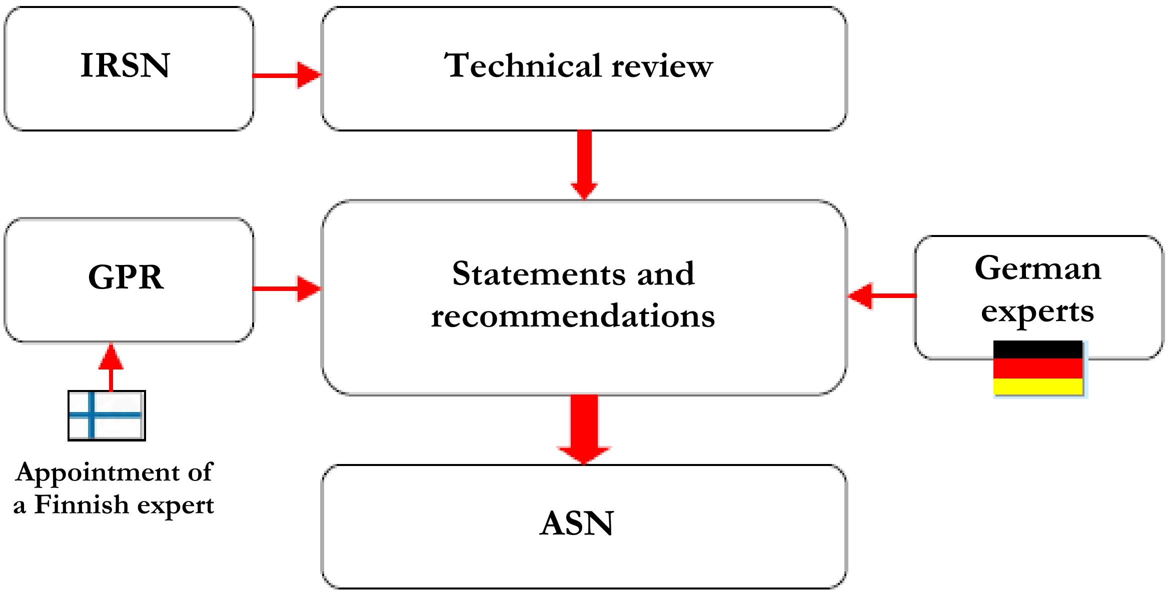



**Figure 3**

During the same period, the review continued on the design and manufacturing choices for large boiler components with three SPN meetings on 17 December 2004, 26 April 2005 and 13 December 2005, respectively.

### *II.3 In 2006*

The SPN met on 6 January 2006 in order to continue the review of design and manufacturing choices for large components.

The GPR and the associated German experts met on 26 January 2006.

In the framework of the technical review in preparation for the authorisation decree, a last meeting of the GPR and the associated German experts is scheduled at the beginning of July. In order to avoid any additional delay, EDF wished that no disagreement on any issue ASN may consider potentially redhibitory among the remaining topics to be reviewed be resolved before the submission of the official request scheduled in early May.

At the end of April 2006, ASN informed EDF and the GPR members of its positions and preliminary requests based on the IRSN's pre-analysis as formalised into an opinion.

On 9 May 2006, EDF submitted its authorisation decree application and sent ASN the official version of the preliminary safety report for "Flamanville 3".

On 29 June and 11 July 2006, the GPR met to review the methodology for assessing the radiological impact of accidents.

On 6 and 11 July 2006, the GPR and the associated German experts met for the last time in order to close the technical-review cycle concerning the safety examination of the EPR at the stage of the preliminary safety report.

On the other hand, ASN and the IRSN both verified the content of the preliminary safety report in relation to the previously reviewed elements concerning the technical requirements intended to be included in the authorisation decree. During the process, a discrepancy between the preliminary safety report and the upstream review was detected and addressed[6].

Lastly, on 30 November 2006, the IRSN sent its opinion on the integration of non-nuclear risks at "Flamanville 3".

---

6. In the version of the preliminary safety report submitted on 9 May 2006 with the authorisation decree application, EDF introduced a change in the previously-review elements relating to the safety criteria for the reference operational situations of the cooling pond.

That issue was detected too late to be presented at the GRP meetings of 6 and 11 July 2006, and was addressed by the IRSN in the opinion it provided in August 2006. The conclusions of the IRSN's opinion, considering that the advocated criteria for the change were not satisfactory, together with the conclusions of the GPR meetings held on 6 and 11 July 2006, were all integrated in a single position letter prepared by ASN, requiring that EDF, if the creation licence was ever issued, to revise those safety criteria by clarifying the orientations to be followed.

## III. Programme of the prerequisite technical review for the authorisation decree

On the basis of the review areas presented in the introduction, the following table summarises the technical topics reviewed by ASN services and their supporting agencies between 2001 and 2006.

*Note:* References (1) to (6) mentioned beside some of the topics appearing below establish a link between the requirements of the position letter on the safety options as described in Section I of this report.

| **Areas of the Review Programme** | | | **Topics reviewed at the stage of the preliminary safety report** | **Review framework** |
|---|---|---|---|---|
| **Nuclear risks** | **General safety objectives** | **Control of the normal operation of the installation** | Radiation protection of workers **(1)** | GPR meetings of 1 July 2004 and 5 July 2005 |
| | | | Radioactive waste **(1)** | GPR meetings of 26 January 2006, 6 July 2006 and 11 July 2006 |
| | | | Chemical and radioactive releases **(1)** | GPR meeting of 26 January 2006 |
| | | **Reduction of the number of significant incidents** | Human-factors integration programme at the design stage **(1)** | GPR meeting of 3 July 2003 |
| | | **Reduction of the core-meltdown risk** | Level-1 probabilistic safety analysis | GPR meetings of 3 July 2003, 6 July 2006 and 11 July 2006 |
| | | | Assessment of the contribution of aggressions to the overall core-meltdown risk | GPR meetings of 6 July 2006 and 11 July 2006 |
| | | | Computerised control and its man-machine interface | GPR meetings of 18 November 2004, 6 July 2006 and 11 July 2006 |
| | | **Practically-eliminated situations** | Heterogeneous-dilution risk of the boron concentration | GPR meeting of 18 November 2004 |
| | | | Global hydrogen-detonation risk in the reactor building | GPR meetings of 1 July 2004 |
| | | | By-pass risk of the containment under severe accident situations | GPR meetings of 5 July 2006, 6 July 2006 and 11 July 2006 |
| | | | Fuel melt in the spent fuel pool | GPR meetings of 3 July 2003 and 26 January 2006 |

| | | | |
|---|---|---|---|
| **Nuclear risks** | **Reduction of the radiological impact of accidents** | Design of the containment building | GPR meetings of 1 July 2004 and 5 July 2005 |
| | | Equipment-hatch of the reactor building | GPR meetings of 1 July 2004 and 1 December 2005 |
| | | Containment of peripheral buildings | GPR meeting of 3 July 2002 |
| | | Core catcher **(4)** | GPR meetings of 18 November 2004, 1 December 2005, 6 July 2006 and 11 July 2006 |
| | | Assessment methodology of radiological impact of accidents | GPR meetings of 29 June 2006 and 11 July 2006 |
| | **Integration of recent experience feedback from operating reactors** | Design basis against internal and external hazard **(5)** | GPR meetings of 18 November 2004, 5 July 2006, 1 December 2005, 26 January 2006, 6 July 2006 and 11 July 2006 |
| | | Sump-clogging risk of ofemergency core cooling systems | GPR meetings of 6 July 2006 and 11 July 2006 |
| | | Crash of commercial aircraft **(6)** | Processing in accordance with defence-secrecy procedure (restricted GPR) |
| | **Innovations introduced in comparison to operating reactors in response to industrial concerns** | Break-preclusion hypothesis: Integration in the safety demonstration of the break preclusion hypothesis of main steam pipes | GPR meeting of 1 December 2005 |
| | | Break-preclusion hypothesis: Validation conditions of the break preclusion hypothesis of primary and secondary pipes **(2)** | GPR meeting of 21 June 2005 |
| | | Principles and approach for equipment qualification | GPR meetings of 5 July 2005 and 26 January 2006, |
| | | Preventive-maintenance operations in power states | GPR meeting of 1 December 2005 |
| | | Digital control and instrumentation**(3)** | GPR meetings of 1 July 2004 and 1 December 2005 |

<table>
<tr><td rowspan="6">Nuclear risks</td><td rowspan="4">Design and manufacturing of nuclear pressure equipment</td><td>Nozzle support ring and the vessel cover</td><td>SPN meetings of 2 July 2003 and 5 January 2006</td></tr>
<tr><td>Envelopes of fuel-bundle control mechanisms</td><td>SPN meeting of 26 April 2005</td></tr>
<tr><td>Pressuriser</td><td>SPN meeting of 17 December 2004</td></tr>
<tr><td>Stream generators</td><td>SPN meeting of 13 December 2005</td></tr>
<tr><td rowspan="2">Miscellaneous</td><td>Design of the security injection system (blocking of pumps and management of “feed-and-bleed” situations)</td><td>GPR meetings of 18 November 2004, 6 July 2006 and 11 July 2006</td></tr>
<tr><td>Modalities to integrate breaches in the reactor-shutdown cooling circuit in the safety demonstration</td><td>GPR meetings of 1 December 2005 and 26 January 2006</td></tr>
<tr><td>Non-nuclear risks</td><td></td><td>Identification and integration of non-nuclear risks in the installation</td><td>IRSN statement of 30 November 2006</td></tr>
</table>

It is important to underline that the technical issues listed in the Table above do not cover in depth the topics addressed in the preliminary safety report, either because:

- the level of review achieved during the examination of safety options was considered sufficient at the stage of the preliminary safety report (e.g., design provisions aiming at eliminating practically all core-meltdown accidents at high pressure), or
- it was considered that those topics constituted neither a sensitive issue with regard to safety objectives nor a fundamentally design-related structuring element, and consequently, that they might be examined, if need be, at the commissioning stage of the reactor (e.g., aeroball-instrumentation system used in German reactors, recruiting and training of agents).

Remark concerning the 60-year operating lifetime mentioned in the creation-licence application, as follows:

> At the EPR-design stage, EDF selected a scenario with a 60-year operating lifetime for sizing specific pieces of equipment, such as the reactor vessel, or for forecasting climate changes (see section on the integration of extreme-heat situations and external-flooding risks).

ASN services feel that it is impossible to decide from an overall standpoint whether a 60-year operating lifetime is valid or not, since the decision relies not only on the arguments presented above, but also on actual alterations to be observed on equipment, the evolution of the industrial sector regarding obsolescence, EDF’s capability to replace equipment when required, and the new safety requirements that ASN may prescribe during the periodic safety review it carries out every 10 years in accordance with French regulations.

Remark concerning the characteristics of nuclear fuel and the corresponding management modalities they imply:

> At the stage of the preliminary safety report, the various proposed methods for managing nuclear fuel correspond to theoretical means with envelope parameters based on actual methods and used to size the installation.

The characteristics of fuel assemblies (design of assemblies, choice of materials, enrichment rate, etc.), as well as the actual management modalities (length of cycles, burnup rate, etc.) would only be examined if an authorisation decree were issued, when time comes to review the application for the initial fuel loading in the reactor (see Section VI).

## IV. Summary of the prerequisite technical review for the authorisation decree

### *IV.1 General safety objectives*

#### *Operational control under normal conditions*

a) Radiation protection of workers

On the basis of a thorough analysis of the operational experience feedback from current reactors, EDF set in place a specific approach to optimise the worksites that contribute the most to the collective dose. With due account of that approach, EDF aims at an average provisional dose of 350 manSv/year over a sequence of shutdowns involving a partial visit, shutdown for simple reloading, a partial visit, shutdown for simple reloading, a partial visit and a decennial visit[7] .

For comparison purposes with an identical shutdown typology, the recorded dose levels within the best reactors in service in France amounts to 440 manSv/year.

b) Waste and releases

When compared to identical fuel management, the EPR reduces by 15% the required quantity of uranium to ensure the generation of a given level of electrical power and, consequently, in a proportion that is about similar to the quantity of waste resulting from spent fuel.

That reduction in uranium consumption is due to the energy-efficiency gains resulting from:

- design choices in the nuclear boiler, accounting for 10% (increase in core size and installation of a neutron reflector in the reactor vessel), and
- the expected improved behaviour of the turbo-alternator, accounting for 5%.

The principles selected for zoning the premises of the installation with a view to controlling the production of technological operational waste are satisfactory. However, the gain assessment on the volume and activity of technological waste or resulting from the process will only be examined after the ultimate delivery of the authorisation decree  on the basis of detailed feasibility studies and in accordance with operating rules to be established. In fact, the design of certain effluent-treatment systems, allows, within the framework of operational activities, to choose between producing waste or proceeding with a release.

#### *Reduction of the number of significant incidents*

Any data relating to system reliability (quality of design and manufacturing) and to the general operating rules will only be examined after the ultimate delivery of the authorisation decree, once contracts for detailed studies will be launched for manufacturing and supplying the required equipment.

With regard to the integration of human factors in the design, both the project structure chosen by EDF in order to take into account that aspect and the work programme committed at the stage of the preliminary safety report are considered to be satisfactory.

7. Known as a VP-ASR-VP-ASR-VP-VD sequence: *VP, visite partielle; ASR, arrêt pour simple rechargement; VD, decennial visit.*

***Reduction of the core-meltdown risk***

a) Level-1 probabilistic safety analysis

Level-1 PSA carried out by EDF at the stage of the preliminary safety report assess the core-meltdown risk at $6.1 \cdot 10^{-7}$ per reactor-year. As a comparison, the risk mentioned in Safety Report VD2[8] of 900-MWe reactors stands at $2.4 \cdot 10^{-5}$ per reactor-year.

That Level-1 PSA was deemed satisfactory at that stage, notably with regard to its use for confirming design choices or for identifying required improvements.

Concerning the result itself, it is important to emphasise that, at that stage, the question only involves a first assessment that will require to be furthered on the basis of a more thorough and more detailed model of the installation and of its operating rules, if the authorisation decree is granted.

b) Assessment of the contribution of aggressions to the overall core-meltdown risk

If the methodology to assess the share of external and internal hazards within the core-meltdown risk is still under development from a probabilistic point of view, the changes brought to the EPR design basis (increase in the severity of some loading cases; design choices, such as geographical separation or "bunkerisation" of specific buildings) reinforce considerably the resistance of the installation to those situations in comparison to current reactors.

c) Computerised C&I and its man-machine interface

One of the lessons learnt from the 1979 accident at the Three Mile Island NPP, in the United States, that caused the partial meltdown of the reactor core, concerns the significance to be given to the ergonomics of man-machine interfaces in the control room.

Consequently, the process to develop man-machine interfaces in computerised regulation of the main control room was examined with special attention being given to the integration of human factors.

The work performed by EDF is currently considered as satisfactory at that stage, particularly with regard to:

- the use of experiments on a simulator with the participation of operators working on operating reactors , and
- the selected orientation to allow more latitude in the decision-making power of operators once again, contrary to the case of N4 reactors, by focusing more the purpose of computerised systems to an assistance mission, outside automated phases.

Since activities involving the development and validation of man-machine interfaces within the computerised C&I of the main control room are not completed yet, it will be important, if the authorisation decree is actually issued, to continue the examination already under way before loading nuclear fuel in the reactor for the first time.

***Practically eliminated situations***

a) Heterogeneous-dilution risk of the boron concentration

An assessment was made of the method with which EDF determined the largest possible volume of clear-water slug that would not compromise the integrity of barriers during its

8. R*apport de sûreté VD2 : mise à jour du rapport de sûreté à l'occasion du réexamen de sûreté effectué lors de la deuxième visite décennale d'une installation* (Safety Report VD2: Update of the Safety Report for the Safety Re-assessment of the Second Decennial Visit to an Installation).

way through  the core. A detailed examination of each heterogeneous-dilution scenario was made after that, by considering the overall lines of defence in place in order to maintain the water volume with an insufficient boron concentration below the critical volume.

As a complement to that deterministic approach, the probabilistic analysis carried out by EDF concerning those sequences was also examined. At that stage in the design, the integration of the heterogeneous-dilution risk was deemed satisfactory. However, the full demonstration of the practically-elimination  of heterogeneous-dilution scenarios will only be achievable after the actual delivery of the authorisation decree, in relation notably to the operating rules that will be set forth and to the update of the probabilistic safety analysis (PSA) relating to those sequences.

b) Overall hydrogen-detonation risk in the reactor building

Preventing the risk of global hydrogen detonation in the reactor building relies on design provisions  (volume of the reactor building and geometry of its internal structures, hydrogen recombiners). The relevancy of those provisions is verified by estimating the quantity of hydrogen produced under severe-accident conditions and by modelling the distribution of the hydrogen concentration within the reactor building.

The assessment of the quantity of hydrogen produced under severe-accident conditions is consistent with current knowledge in the field. On the other hand, the distribution of hydrogen concentration within the reactor building requires further investigations, if an actual authorisation decree is issued, notably regarding the impact of the "two-room" concept (series of radiological containment and radiation protection features implemented within the reactor building with a view to carrying out maintenance activities while in power state), since that concept may generate various localised hydrogen concentrations that are not homogeneous.

c) Risk of the containment bypass under severe-accident conditions

In the states where the equipment-hatch of the reactor building is open, the possibility to close the containment within a two-hour delay is deemed compatible with the timeframe of core-meltdown accidents that are not considered as practically eliminated.

Concerning containment-bypass risks via circuits connected to the primary circuit, a deterministic analysis of each identified scenario and of the associated lines of defence was conducted and deemed satisfactory[9]. However, at that stage, the supporting PSA does not allow to determine whether those sequences are practically eliminated or not: a full demonstration of the practically-elimination of containment-bypass scenarios under severe-accident conditions will only be achieved after the actual delivery of the authorisation decree, in relation notably to the update of the PSA for those sequences.

d) Fuel meltdown in the spent fuel pool

With respect to the reliability of the water-cooling function in the spent fuel pool, the Level-1 PSA results were used to detect the excessive sensitivity of the initial design to a common-mode failure involving the two redundant systems that were originally planned for that function. The selected design evolution consisting in adding a third diversified cooling system was reviewed and approved.

9. If the deterministic analysis performed by the IRSN for the GPR meeting of 5 July 2005 helped to assess the design as satisfactory, the IRSN noticed during the further probabilistic analysis conducted for GPR meetings of 6 and 11 July 2006, that EDF introduced without prior warning a design change that contradicted the positions taken in 2005. Following the GPR meeting, ASN sent a position letter to EDF requiring that the change be cancelled.

With regard to the drainage of the spent fuel pool, design specifications were established in order to ensure that any leak or breach on a circuit connected to the spent fuel pool be either excluded or unable to cause the direct uncovering of the stored fuel assemblies, even in the absence of any isolation measures.

It is important to underline that the drainage-risk analysis did not take into consideration the scenario involving the break of a isolation plug of a steam generator during maintenance activities, with due account of operating rules planned by EDF. The installation of those plugs should only be allowed once the core content is fully unloaded in the spent fuel pool and a double isolation has been implemented between the reactor building's pool and spent fuel pool.

In drainage scenarios that avoid the direct uncovering of fuel assemblies, but in which it is impossible to maintain a sufficient water level to maintain the suction of the spent fuel pool's cooling system, an emergency make-up system first prevents the delayed uncovering of fuel assemblies by water boiling and secondrestore a sufficient water level in order to restart the cooling system.

As a further complement to that deterministic approach, which is deemed satisfactory at the preliminary safety-report stage, the full demonstration of the practically-elimination of fuel-meltdown scenarios in the spent fuel pool shall only be achievable once an actual authorisation decree has been issued, in relation notably to the update of the sequence-related PSA.

***Reduction of the radiological impact of accidents***

a) Design of containment building

With regard to the reference solution in 2000, EDF has decided to replace the composite liner of the internal face of the first wall of the containment system by a metallic liner.

In addition, EDF used a new approach to size the containment by separating the following two notions:

- the design pressure of the containment , set at 5.5 bars (absolute), associated with the pressure that the containment may withstand without any functional or structural consequence, and
- the leaktightness verification pressure , set at 6.5 bars (absolute), for which adapted design criteria ensure the leaktightness of the containment , but without excluding structural consequences on it (e.g., permanent deformation of the leak-proof metallic liner).

In the safety-demonstration studies for that reactor containment building, EDF distinguished between the following:

- scenarios of so-called "representative" accident conditions, for which it was confirmed that the resulting pressure was lower than the design pressure of the containment , and
- scenarios of so-called "limit" accident conditions, which relate to a lower probability of occurrence, and for which the design pressure may be exceeded, while remaining lower than the leaktightness verification pressure.

That approach is not strictly consistent with recommendations contained in the technical design and construction guidelines for the next generation of PWRs, in which references were only made to a design pressure without any probabilistic considerations on scenario selection. However, the newly proposed design and the associated safety approach were altogether deemed satisfactory, because they offered more margin in maintaining leaktightness under high-pressure conditions.

Nevertheless, if an actual authorisation decree is issued, it will be necessary to further the demonstration of long-standing containment integrity against "limit" accident scenarios by ensuring notably that the global deflagration of the maximum quantity of hydrogen contained in the reactor building under severe-accident conditions generates lower pressures than the leaktightness verification pressure.

b) Equipment-hatch of the reactor building

In light of the peculiarity of the equipment hatch, special attention was given to the design of that part of the reactor containment building. The assessment performed at that stage helped to ascertain that the selected design evolution principles were satisfactory to account for the difficulties encountered with operating reactors .

However, if an authorisation decree is issued, it will be necessary to perform a further thorough examination of the detailed design studies on the equipment-hatch and the other peculiarities of the reactor containment building.

c) Containment of peripheral buildings

With the EPR, peripheral buildings contribute to the containment of radioactive materials by being involved in collection and filtration operations before discharging any potential leakage from the reactor building s penetrations and openings.

EDF's approach and selected scenarios to determine the effectiveness of the containment of peripheral buildings were examined and deemed satisfactory on the basis of currently available design data.

d) Core catcher

Each of the following key operating phases of the core catcher was examined in detail:

- the temporary retention in the reactor pit;
- the corium transfer conditions from the reactor pit to the spreading room;
- the risk of under water molten-corium pouring in the spreading room; and
- the retention and cooling of corium in the spreading room.

Thanks to that examination, it was possible to improve some of the technical features proposed by EDF, and notably to increase the thickness of the sacrificial concrete in the spreading room or to use a fusible aluminium plate supported by a steel grid for the gate of the transfer channel from the reactor pit towards the spreading room.

On the basis of available experimental results and of the review of the representativeness of the selected sizing scenarios, the robustness of the core catcher design was deemed satisfactory.

e) Assessment methodology for the radiological impact of accidents

While EDF was developing in parallel two accident-assessment methodologies, one for its current operating reactors , and the other specifically for the EPR through discussions with its German partners, ASN addressed the relevancy of such approach in 2004.

Interrogated by ASN on the appropriateness of the situation, EDF decided to orient its work on developing a single assessment methodology resulting from the convergence of both approaches.

With due account of the required time for developing and reviewing the new methodology, ASN accepted EDF's proposal to present the following items in the preliminary safety report of the EPR:

- the results of the initially planned methodology;
- the newly proposed methodology to be applied, if an actual authorisation decree is issued, in the framework of the update of the prerequisite safety report to be submitted before any fuel is loaded for the first time in the reactor, and
- a sensitivity analysis in order to assess immediately any influence on the results submitted of any scenario or parameter changes introduced by the new methodology.

The scope of the sensitivity analysis was validated by ASN on the basis of IRSN technical recommendations. The new methodology (not addressing severe accident situations), common to both current reactors and the EPR, was the subject of a GPR statement and recommendations. A complementary GPR review of the part of the methodological addressing severe accident situations is scheduled in 2008.

Although the assessment results of the radiological consequences of an approved methodology are not available yet, ASN services feel that the overall existing design requirements for the EPR should meet the prescribed radiological objectives. Beyond that qualitative judgement, it will be necessary, if an actual authorisation decree is issued, to verify quantitatively the performance of those specifications before loading fuel assemblies in the reactor for the first time.

### *IV.2 Integration of experience feedback from operating reactors*

#### *Design reference systems for external and internal aggressions*

After examining design basis with regard to various hazards, such as internal fires, internal explosions, lightning, extreme cold or hot weather conditions, earthquakes, external flooding, relevant combinations of hazards to be taken into account, as well as the design of the pumping station in relation to the risk of total loss of the main heat sink , no stumbling block was identified for the delivery of the authorisation decree.

The final conclusions about extreme-hot weather conditions and external-flooding risks, which correspond to recent events, are detailed as examples below.

a) Extreme-hot weather conditions

Concerning extreme-hot weather conditions, EDF decided to reinforce the facility's design compared to the selected approach for existing reactors. The purpose is not only to verify that such situations do not compromise the safety of the installation, but also to take them into account in the initial sizing as part of normal operating conditions.

As input parameters for sizing the Flamanville-3 reactor, EDF set forth maximum temperature values for air and for the water of the English Channel, by integrating an average-temperature evolution scenario until the end of the 21$^{st}$ century and by adding to those average temperatures a peak corresponding to random fluctuations with an occurrence probability of once every 100 years.

Submitted by EDF to the experts of *Météo-France,* of the Dynamic Meteorology Laboratory (*Laboratoire de météorologie dynamique* – LMD) of the Pierre-Simon-Laplace Institute and of

Orsay University, the validity of the studies conducted on climate and air-temperature evolutions was not questioned.

The selected sizing values for Flamanville have been set as follows:

- for air: a maximum average daily temperature of 36°C and a maximum instantaneous temperature of 42°C, and
- for sea water (English Channel): a maximum daily temperature of 26°C.

Such an approach integrating extreme-hot weather conditions at the design stage is deemed satisfactory and the selected temperature values are also deemed relevant in light of current predictive models. However, since uncertainties remain about climate evolutions until the end of the century, ASN requested EDF to forecast, as a complement, the possibility to adapt the installation to any actual climate changes that would prove more conservative than current forecasts.

b) Risk of external flooding

In accordance with *Basic Safety Rule I.2.e* (*Règle fondamentale de sûreté* – RFS) concerning the integration of external-flooding risks, the final specifications of the installation were set above the maximum design flood level (*cote majorée de sécurité* – CMS), which corresponds to the accumulation of a tidal coefficient of 120 and a millennial sea surge. As a complement, EDF has already taken into account the progress achieved in the work under way for the revision of RFS I.2.e, which integrates the event that occurred on the Blayais Site during the December 1999 storm. Consequently, the installation-design approach integrated a protection provision for the required equipment to perform safety functions in case of complementary contingencies (waves, rain, etc.) and their combination. Lastly, protective means against combinations of contingencies involving CMS were also determined by integrating an additional margin in order to counter likely climate evolutions over the medium term. However, due to the uncertainties concerning the assessment of the actual margin provided over the long term by the platform of the pumping station set at 0.75-m above the current maximum design flood level (the platform of the nuclear island being set at 4.6 m above that level), that issue will need to be followed up through the periodical safety review, if an actual authorisation decree is issued.

***Sump clogging risk of emergency core cooling systems***

A risk was detected recently in operating reactors concerning sump-clogging of emergency core cooling systems due to the debris generated under accident conditions in the catch basins located within the reactor building.

In the framework of the EPR Project, EDF has selected the following approach in order to ensure the sound operation of emergency cooling systems under accident conditions:

- preventing as much as possible any risk-inducing factors involving the clogging of RIS[10] and CHRS[11] emergency cooling systems in the IRWST[12] by paying special attention to the selection of the materials to be used in the reactor building, notably with regard to thermal insulation;
- limiting the quantity of resulting debris transferred to the IRWST , notably with regard to the emplacement of curbs, screens and retention baskets, and

10 RIS: reactor water-injection system in the reactor under accident conditions.

11 CHRS: containment heat removal system under accident conditions involving core meltdown (*évacuation ultime de la chaleur* – EVU).

12 IRWST: in-containment refuelling water storage tank; once open, that tank of borated water located in the lower part of the reactor also serves as a sump.

- ensuring the protection of RIS and EVU pumps against the debris carried away by the IRWST fluid by installing immersed filter cages. The filtering surface of RIS filters is sized on the basis of the estimated term source of debris for accidents without core meltdown. However, from the standpoint of in-depth defence, an active declogging system is planned in the design. With regard to the EVU system, since no envelope estimate of the nature and quantity of resulting debris involved under core-meltdown conditions is available due to the current state of knowledge, the protection of pumps is guaranteed by the largest filtering surface possible, with due account of the available space, and an active declogging system.

After review, that approach was deemed satisfactory:

- against accident situations without core meltdown, and
- at the current stage, against accident situations with core meltdown, in light of current information on physical and chemical phenomena involved in such situations.

If an actual authorisation decree is ever issued for Flamanville-3, it will be necessary to further the review of the detailed design of those overall lines of defence and to take stock on the advances in the knowledge on that topic before loading any fuel in the reactor for the first time. Since that issue also concerns existing reactors, ASN ensures that EDF will continue its characterisation work on the debris' source term and on the clogging phenomena associated with severe-accident conditions, independently from the licensing procedure for the creation of Flamanville-3.

### *IV.3 Innovations compared to operating reactors in response to industrial concerns*

***Break-preclusion hypothesis***

"Break exclusion" applies to any circumferential double-ended pipe breaks.

a) Integration of the break-preclusion hypothesis involving main steam pipes in the safety demonstration[13]

Although EDF selected the scenario excluding a guillotine break of the main steam lines within the reactor building and beyond the tappings of relief valves and of the main steam isolation valve outside the reactor building, the guillotine break of a main steam pipe coming out of the steam generator has been maintained in the list of Category-4 reference accident conditions for the purpose of the authorisation decree application. In its current state, the sizing of the installation with regard to steam-pipe ruptures is deemed satisfactory.

However, if an actual authorisation decree is issued, EDF has announced that it intends to replace ***as Category-4***, in the preliminary safety report it would submit before loading any fuel in the reactor for the first time, the study on the guillotine break of a main steam pipe coming out of the steam generator ***by a study*** of the scenarios integrating the selected break-exclusion hypothesis.

Without being opposed to such perspective, ASN has already pointed out to EDF that, from the defence-in-depth standpoint, the guillotine break of a main steam pipe within the reactor building should be maintained together with realistic hypotheses in the framework

[13] The integration of the break-preclusion hypothesis involving primary pipes in the safety demonstration was reviewed at the safety-option stage.

of the sizing of the containment envelope and of the qualification of equipments located within the reactor building.

b) Validation conditions of the break-preclusion hypothesis involving primary and secondary pipes

ASN's review helped to set the objective of the break-preclusion demonstration as guaranteeing that pipe integrity will be maintained throughout the lifetime of the installation. Integrity is meant as the absence of any equipment degradation that would compromise damage prevention.

After the review, ASN specified that the demonstration may be used if the assessment of the technical provisions implemented during the design, manufacturing and operating stages leads to the conviction that any pipe break is highly unlikely.

ASN considers that the break-preclusion hypotheis pertains to the first level of a "defence-in-depth" safety approach. That level consists of guaranteeing the quality of design, manufacturing and follow-up during service, with the quality guaranty for design and manufacturing being based jointly on the quality of rules being enforced, the verification of that enforcement and the final control of those activities.

Among the elements presented by EDF, design and manufacturing choices rely on the application of the Level-1 Design and Construction Rules for the Mechanical Components of PWR Nuclear Islands (*Règles de conception et de construction applicables aux matériels mécaniques des réacteurs nucléaires à eau sous pression* – Code RCC-M) for all main-pipe components included in primary and secondary circuits. ASN felt that such provisions should favour the successful break-preclusion demonstration provided that the process-certification requirements are extended to secondary pipes. On the other hand, ASN feels that, among the elements presented by EDF, design verification must be reinforced, notably with regard to the validity of data and to loading-related scenarios. In addition, provisions for in-service inspections were deemed unsatisfactory.

After the review, the operator submitted a report on the evolution of the case, with due account of the essential principles specified by ASN. Those new elements, except for those relating to in-service inspections, appear in the preliminary safety report.

If an actual authorisation decree is issued, EDF's programme for in-service inspections regarding the circuits involved in the break-preclusion hypothesis will require to be reviewed before any fuel is loaded in the reactor for the first time.

### *Equipment-qualification principles and approach*

In order to fulfil the equipment-qualification requirement under accident conditions, EDF has proposed a different approach than the current one for operating reactors .

According to the new approach, equipment qualification under accident conditions within the reactor building does not rely on a single standard, but on several standards, which have been set forth in accordance with an analysis of functional requirements for equipment, not only in terms of ambient conditions (pressure, temperature, hygrometry, radiation, etc.), but also in terms of mission timescale.

The review made on that aspect dealt firstly with defining the different standards being contemplated and the associated qualification profiles. It led EDF to revise its initial proposal. Once the standards were defined and the associated qualification profile was deemed satisfactory, the review addressed the principles to identify equipments to be qualified, their relevant functional requirements, as well as the methodology for selecting their corresponding qualification standard. Concerning both aspects, even if the approach submitted by EDF is deemed satisfactory at the current stage, that judgement will need to be confirmed by a more thorough review before any fuel is loaded in the reactor for the fist time, especially with regard to the instrumentation-qualification strategy in relation to the expected operating robustness under accident conditions.

In addition, it was deemed acceptable to use other qualification methods than the one developed in France, provided that EDF assesses thoroughly the adequacy of the corresponding qualification specifications to the specific data of the EPR.

***Preventive-maintenance operations in power states***

Concerning the possibility envisaged by EDF to perform certain preventive-maintenance operations in power states in order to reduce outages duration, the review conducted at the current stage has consisted in defining the reference safety-related conditions when conducting such operations. The conditions selected by EDF after the review were deemed satisfactory.

If an actual authorisation decree is issued for Flamanville-3, it will be necessary, before loading any fuel in the reactor for the fist time, to fulfil those requirements in the various operating aspects designed by EDF through the full list of relevant systems and the details of the associated preventive-maintenance operations.

***Digital instrumentation and control systems***

At that design stage, the review dealt with:

- the general architecture of instrumentation and control systems with special attention to the architecture of the systems performing F1A1 functions[14] and to the remote shutdown station, and
- EDF's decision to implement instrumentation and control systems for performing F1A functions by relying on AREVA-NP's Teleperm-XS industrial programmable and digital platform.

With regard to the overall architecture of I&C systems, if an actual authorisation decree was issued, important issues such as the diversity of the equipment to be selected, the use of networks between the various I&C equipments and the use of programmed components will need to be thoroughly reviewed before any fuel is loaded in the reactor for the first time.

Concerning EDF's decision to implement instrumentation and control systems for performing F1A functions by relying on AREVA-NP's Teleperm XS industrial programmable and digital platform, a pre-examination of the opportunity and conditions of furthering the safety assessment of that platform was needed due to the complexity level of such type of system.

During that first step, it was therefore possible to ensure that the level of accessible information for the IRSN's assessment was sufficient (access to source codes, French translation of documentation initially only available in German, etc.). It also made it possible to verify that EDF's choices concerning the selected sub-assembly for Flamanville-3 in relation to the overall options provided by the Teleperm-XS platform (exclusion of certain configurations, exclusive use of compatible programmes with "white-box" requirements for F1A functions, etc.), actually removed the IRSN's initial reservations.

[14] Security classification F1A corresponds to the security functions required for achieving a controlled state after the occurrence of a Category-2 to Category-4 reference incident and accident situations.

Later, a more thorough review of that technological solution took place with a special attention being given to the development of computerised safety systems based on the Teleperm-XS platform (cycle of software development, tests and analysis of the source code).

After the review, it was estimated that the sub-assembly of the selected Teleperm-XS platform for Flamanville-3 constituted an acceptable basis for the development of the protection system.

In general, if an actual authorisation decree is issued, it will necessary to pay special attention to a detailed assessment and to the control of the development of digital I&C systems before any fuel is loaded in the reactor for the first time.

### IV.4 Design and manufacturing of nuclear pressure equipments

#### Nozzle support ring and reactor vessel head

The Technical Rules relating to the construction of the future main primary and secondary circuits in PWRs (Reference [47]), which have been integrated in the Order of 12 December 2005 concerning nuclear pressure equipments, require that the number of welds be as low as possible at the design stage. The purpose of such requirement is to limit the zones where deficiencies may occur in the metal and, consequently, constitute potential crack sources, while reducing professional doses by the welding-control staff. The EPR-vessel constitutes an improvement in that regard, since it consists of a nozzle support ring that integrates the clamp on which the operating cover will be assembled.

The selection of a monoblock ring requires the blacksmith to work from a full and heavy ingot rather than from a hollow and relatively light one, as used for N4 vessels. ASN insisted on requesting that the control of manufacturing operations by the contracted blacksmith, Japan Steel Works, be demonstrated, especially with regard to the reproducibility of the ring-quenching operation, which ensures the required quality of the mechanical characteristics. Since then, EDF and the manufacturer of the nozzle support ring have provided the expected demonstration elements.

The study of the  noin ductile failure of the vessel head has led ASN to seek a justification for the validity of the selected value of -30°C for the ductile/fragile transition temperature in the constituting metal of the welding joint between the cap and the clamp. In addition, as in the case of the nozzle support ring, ageing and fatigue studies must justify both the absence of ageing modes and the validity of the selected conservative deficiency. Lastly, in connection with expected complementary justifications, ASN has requested that the industrial feasibility of a monoblock vessel head be investigated, as in the case of 900-MWe reactor vessel head, in order to ascertain whether the welded joint between the cap and the clamp should be discarded or not.

Furthermore, in case of accidental breach in the primary circuit, the core dewatering may lead to its meltdown. In order to prevent that risk, the distance between the axes of the pipes and the top part of the active core was increased by one-third.

Lastly, in order to limit the consequences of a hypothetical core meltdown, the bottom of the vessel is free of instrumentation penetrations, which are fitted rather in the vessel head.

In conclusion, the design choices for the reactor vessel were deemed acceptable by ASN at the current stage. However, if an actual authorisation decree is issued, it will be necessary to further the studies aiming at justifying the mechanical strength of the nozzle support ring and of the vessel head against ageing and fatigue.

#### Control rod drive mechanisms

In comparison to EDF's operating reactors , the presence of a core catcher on the EPR leads to a new architecture relying exclusively on the reactor vessel head for fitting all required core-instrumentation penetrations.

For the pressure envelopes of control rod drive mechanisms, the decision was made to select a design that was close to the current one already fitted on German KONVOI-type reactors, which integrate the core-instrumentation penetrations in the vessel head. ASN noted that the design changes for those pressure envelopes did not constitute an improvement to the N4 design, given the presence of four strength welds per pressure envelope and the sensitivity of the constituting materials of those envelopes. That is the reason why ASN requested that an adapted monitoring system be established and justified in order to take into account the presence of those welds and of the materials involved.

Guarantees will also need to be provided by the manufacturer about the supply, assembly and ageing of those materials. ASN has also requested that suitable devices be designed to limit oscillations at the extremities of the pressure envelopes of the control rod drive mechanisms in case of earthquake.

For all applications, every additional complement will need to be examined, if an actual authorisation decree is ever issued.

***Pressurizer***

As in the case of the reactor vessel, the EPR pressurizer has fewer welds than the N4 pressurizer. N4 rings consist of several formed and welded steel sheets; the three EPR rings will be cast. ASN also feels that the following provisions are satisfactory:

- the reduction of impurity concentration in the selected steel;
- the separation between normal and auxiliary spraying nozzles;
- the improved accessibility to the bottom of the pressuriser, and
- the best maintenance capability of the spraying nozzles and of anti-condensation heaters.

In conclusion, the design choices for the pressurizer were deemed acceptable by ASN.

***Steam generators***

The design of EPR steam generators is close to that of N4 reactors. That choice is an integral part of the general approach consisting in selecting proven technical solutions for the EPR, without excluding some evolutions. The review has shown that the design choices for EPR steam generators should not be questioned in relation to the reference system used for the review. However, if an actual authorisation decree is issued, EDF will need to provide some justifications once more, especially concerning the selected specifications for protecting the equipment against overpressure.

### *IV.5 Non-nuclear risks*

In its assessment of non-nuclear risks at Flamanville-3, EDF relied on the applicable approach for classified installations for environmental purposes (*installation classée pour la protection de l'environnement* – ICPE). That approach is deemed satisfactory and the results of its application raise no non-nuclear risk for populations and the environment.

## V. Conclusion of ASN services concerning the authorisation decree application

ASN services emphasise the fact that Flamanville-3 was submitted to a much larger and thorough review than previous French nuclear power reactors at the stage of the preliminary safety report.

**On the basis of the available information at the current stage and with due account of the progress report on technical instructions performed, ASN services:**

1) **have raised no issue concerning compliance with general safety objectives;**
2) **consider that integrating the acquired experience on operating reactors since the approval of EPR safety options is satisfactory;**
3) **consider that the new changes compared to the design of operating reactors in response to industrial concerns are acceptable from a safety standpoint;**
4) **are not questioning at the current stage of the project the overall design choices for the large components of the main primary and secondary circuits, and**
5) **have detected no significant non-radiological industrial risk for the populations and the environment.**

**In conclusion, ASN services have identified no technical argument against the delivery of an authorisation decree for the Flamanville-3 INB.**

## VI. Continuation of the safety assessment beyond the authorisation decree

Since an authorisation decree  might be issued, the safety assessment of the Flamanville-3 INB by ASN services and their technical support agencies should be pursued before any fuel is actually loaded in the reactor for the first time.

In relation to the review carried out before the delivery of the authorisation decree, which is based essentially on the assessment of design basis , that new review phase will be extended to construction and manufacturing conformity and to the review of the general technical and organisational operating rules.

**Creation-licence decree**

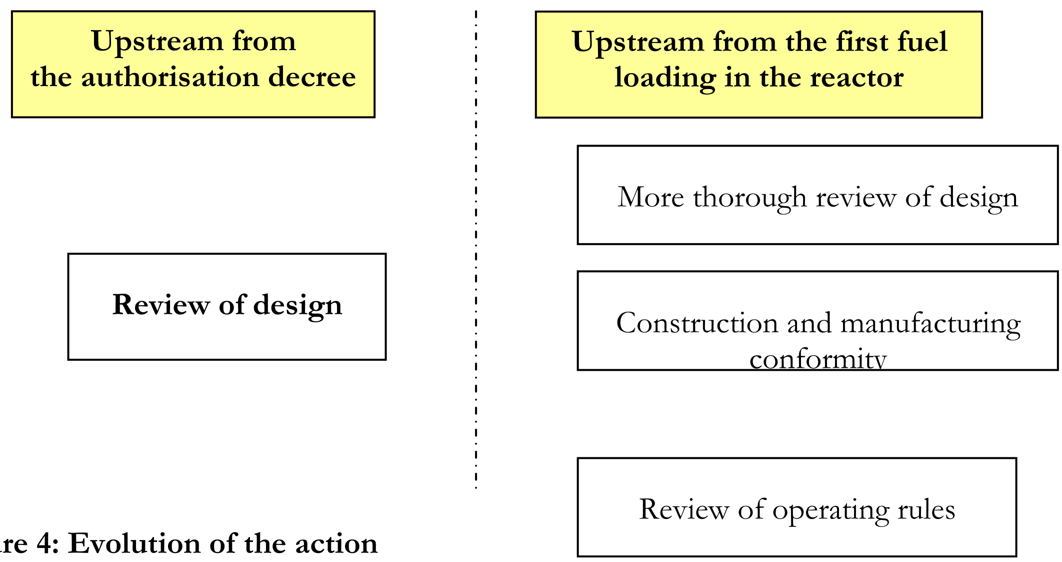


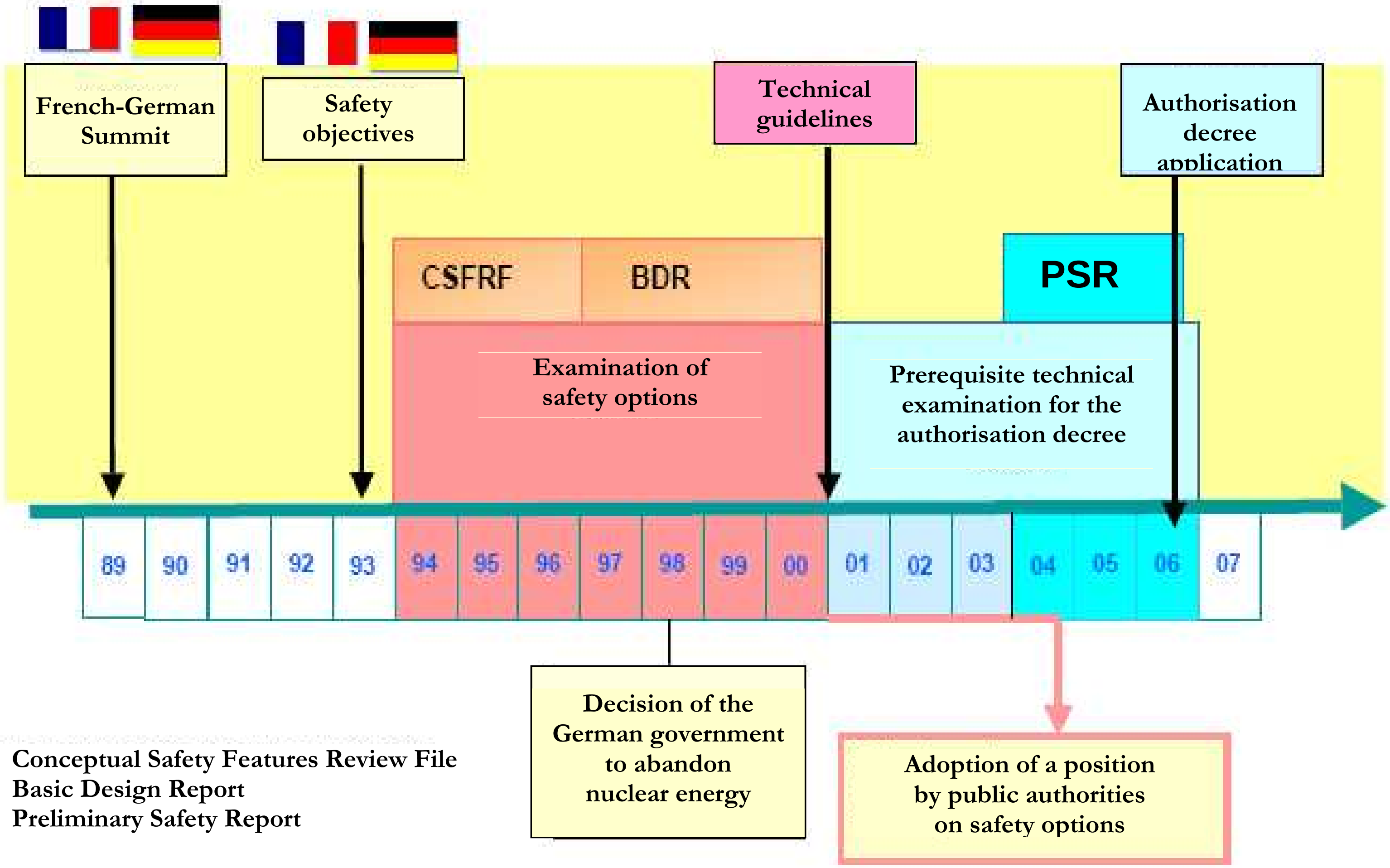

**Figure 4: Evolution of the action of ASN services beyond the authorisation decree application**

Besides continuing the review of the technical aspects mentioned in this report, other complementary topics will need to be examined, such as the operating principles and rules of the installation or the nuclear fuel management .

The development of the detailed review programme is under way with the support of the IRSN. If an actual authorisation decree is issued, the objective is to finalise the programme before 31 March 2007.

ASN services, together with the IRSN's support, are working at implementing a construction-control programme. If an actual authorisation decree is issued, an inspection programme for 2007 is now ready to be launched. The first inspections would deal first with quality management and supplier control, and second, with civil-engineering construction activities. In addition, for the nuclear pressure equipment constituting the primary and secondary circuits, ASN services will ensure that their design and manufacturing are consistent with the requirements prescribed by the Order of 12 December 2005.

*Annex I*
**General chronology of the EPR review**




| | |
|---|---|
| CSFRF | Conceptual Safety Features Review File |
| BDR | Basic Design Report |
| PSR | Preliminary Safety Report |

## *Annex II*
## Detailed presentation of the review process with GPR consultation

**Gauging meeting**

At the meeting, ASN shall inform EDF of the topics to be reviewed;

The IRSN and EDF shall take stock of the technical documents that have already been submitted or are to be submitted promptly by EDF, in support of the IRSN analysis.

After the meeting, ASN shall formalise the review framework in a formal letter sent to the GPR.

**Technical review**

IRSN experts shall conducted the technical review on the basis of:

– the technical documents submitted by EDF after the scoping meeting;
– technical meetings, and
– questionnaires being sent to EDF which, in turn, shall formalise its replies.

**IRSN Report**

The summary report of the review to be presented to the GPR members shall be prepared by the IRSN. The summary report shall:

– reference all documents being analysed;
– describe EDF's approach and positions, and
– present the IRSN analysis and conclusions.

**Preparatory meeting**

At the meeting, the IRSN shall present its report project to EDF in the presence of ASN and of the GPR members who wish to attend. EDF shall have an opportunity to formulate its objections or its agreement in relation to the IRSN's conclusions.

After the meeting, EDF's proposed position and action shall be formalised by a letter preceding the plenary meeting.

**GPR meeting**

At the GPR's plenary meeting, the IRSN shall present its report and conclusions, EDF's proposed positions and actions, as well as the recommendations submitted to the GPR. EDF shall attend and may defend its positions in case of disagreement with the IRSN's proposed recommendations. The meeting shall be concluded by the preparation of the GPR's opinion and recommendations to be sent to ASN. Those documents are referenced in this report.

**EDF letter formalising positions and actions**

**GPR opinion and recommendations**

**ASN positions and requests**

Documents referenced in this report

*Annex III*

**References**


***Summary of the safety-option review***

[1] Position letter of public authorities on the safety options of the EPR Project DGSNR/SD2/ No. 0729/2004 of 28 September 2004.

***GPR meeting of 3 July 2002***

[2] IRSN-GRS joint report No. 83.
[3] GPR statement and recommendations No. 02-24 of 22 July 2002.
[4] ASN position letter: DGSNR/SD2/No. 240-2003 of 9 April 2003.

***GPR meeting of 3 July 2003***

[5] IRSN-GRS joint reports No. 84-85-86.
[6] Confirmation of EDF's positions and actions: ECEM03.0097 of 22 July 2003.
[7] GPR statement and recommendations No. 03-25 of 31 July 2003.
[8] ASN position letter: DGSNR/SD2/No. 132-2004 of 23 February 2004.

***GPR meeting of 1 July 2004***

[9] DSR report No. 18 (EPR report No. 87).
[10] Confirmation of EDF's positions and actions: ECMT.040019 of 9 July 2004.
[11] GPR statement and recommendations No. 04-12 of 20 July 2004.
[12] ASN position letter: DGSNR/SD2/No. 640-2004 of 23 August 2004.

***GPR meeting of 18 November 2004***

[13] DSR report No. 34 (EPR report No. 88).
[14] Confirmation of EDF's positions and actions: ECMT040059 of 2 December 2004.
[15] GPR statement and recommendations No. 04-23 of 20 December 2004.
[16] ASN position letter: DGSNR/SD2/No. 0181-2005 of 14 April 2005.

***GPR meeting of 5 July 2005***

[17] DSR report No. 69 (EPR report No. 89).
[18] Confirmation of EDF's positions and actions: ECMT050085 of 11 July 2005.
[19] GPR statement and recommendations No. 05-23 of 18 July 2005.
[20] ASN position letter: DEP/SD2/No. 0440-2005 of 10 August 2005.

***GPR meeting of 1 December 2005***

[21] DSR report No. 69 (EPR report No. 89) and No. 92 (EPR report No. 90).
[22] Confirmation of EDF's positions and actions: ECMT050144 of 13 December 2005.
[23] GPR statement and recommendations No. 05-35 of 16 December 2005.
[24] ASN position letter: DEP/SD2/No. 0171-2006 of 27 March 2006.

***GPR meeting of 26 January 2006***

[25] DSR report No. 103 (EPR report No. 91).
[26] Confirmation of EDF's positions and actions ECMT060045 of 28 February 2006.
[27] GPR statement and recommendations No. 06-06 of 6 March 2006.
[28] ASN position letter: DEP/SD2/No. 0197-2006 of 6 April 2006.

***Processing of the "radiological impact" file and GPR meetings of 29 June and 11 July 2006***
[29] Letter DGSNR/SD2/No. 0424-2004 of 26 May 2004.
[30] Letter DEP/SD2/No. 2081-2004 of 6 January 2005.
[31] DSR statement and recommendations No. 2005-327 of 26 August 2005.
[32] Letter DGSNR/SD2/No. 597-2005 of 26 December 2005.
[33] DSR report No. 127.
[34] Confirmation of EDF's positions and actions: ENSN060089 of 27 September 2006.
[35] GPR statement and recommendations No. 06-27 of 13 October 2006.

***GPR meetings of 6 and 11 July 2006***
[36] DSR report No. 128 (EPR report No. 92).
[37] IRSN statement and recommendations of preliminary analysis: DSR No. 2006-131 of 6 April 2006.
[38] ASN position letter: DGSNR/SD2/ No. 0236-2006 of 28 April 2006.
[39] EDF response: ECEP060695 of 21 June 2006.
[40] GPR statement and recommendations No. 06-21 of 19 July 2006.
[41] Confirmation of EDF's positions and actions: ECMT060119 of 24 July 2006.
[42] IRSN statement and recommendations DSR-No. 2006-306 of 7 August 2006 mentioning disagreement on the content of the preliminary safety report.
[43] ASN position letter: DEP-DCN/ No. 025-2007 of 25 January 2006.

***IRSN opinion on non-nuclear risks***
[44] IRSN statement and recommendations: DSR-2006-379 of 30 October 2006.
[45] ASN position letter: DEP-DCN/No. 0641-2006 of 19 January 2006.

* * *

***Framework of the review of nuclear pressure equipment***
[46] DGSNR-GRE/BCCN/DE/AR No. 020252 of 10 June 2002.
[47] Letter DSIN/GRE/BCCN/99081 of 19 October 1999.

***SPN meeting of 2 July 2003***
[48] SPN statement and recommendations: FC/MA No. 030320 of 18 July 2003.
[49] Letter DGSNR/SD5/FC/MFG No. 030465 of 15 October 2003.

***SPN meeting of 17 December 2004***
[50] SPN statement and recommendations: FC/MFG No. 04162 of 17 December 2004.
[51] Letter DGSNR/SD5/FC/MFG No. 041694 of 22 January 2005.

***SPN meeting of 26 April 2005***
[52] SPN statement and recommendations of 26 April 2005.
[53] Letter DGSNR/SD5/PM/MFG No. 050258 of 06 May 2005.

***SPN meeting of 21 June 2005***
[54] SPN statement and recommendations: FC/MA No. 050602 of 21 June 2005.
[55] ASN position letter: DEP/SD5/No. 0074-2006 of 13 February 2006.

***SPN meeting of 13 December 2005***
[56] SPN statement and recommendations: PM/MFG No. 05-615 of 7 March 2006.
[57] Letter DGSNR/SD5/PM/MFG No. 0100-2006 of 20 March 2006.

***SPN meeting of 5 January 2006***
[58] SPN statement and recommendations: FC/AR DEP-SD5-0044-2006 of 24 February 2006.
[59] Letter FC/VF DEP-SD5-0181-2006 of 18 May 2006.

*Annex IV*

## Reminder about safety objectives set forth by ASN

**(Objective 1) Controlling the normal operation of the installation**

For normal operation and abnormal occurrences, one objective is the reduction of individual and collective doses for the workers, which are largely linked to maintenance and in-service inspection activities. Reduction of the occupational exposures shall be aimed at by an optimization process taking into account the data obtained from operating experience. Consideration must also be given to the limitation of radioactive releases within the corresponding dose constraints, and to the reduction of quantities and activities of radioactive wastes.

**(Objective 2) Reducing the number of significant incidents**

Another objective is to reduce the number of significant incidents, which involves seeking improvements of the equipment and systems used in normal operation, with a view to reducing the frequencies of transients and incidents and hence to limiting the possibilities of accident situations developing from such events.

**(Objective 3) Reducing the core-melt frequency**

A significant reduction of the global core melt frequency must be achieved for the nuclear power plants of the next generation. Implementation of improvements in the "defence-in-depth" of such plants should lead to the achievement of a global frequency of core melt of less that $10^{-5}$ per plant operating year, uncertainties and all types of failures and hazards being taken into account.

**(Objective 4) Reducing the radiological impact of accidents**

In addition, a significant objective is to achieve a significant reduction of potential radioactive releases resulting from all conceivable accident situations, including core-melt accidents. For accident conditions without core melt, there shall be no need for protective measure for people  living in the vicinity of the damaged NPP (neither evacuation, nor sheltering).

Accident situations with core melt which would lead to large early releases have to be "practically eliminated" : if they cannot be considered as physically impossible, design provisions have to be taken to design them out. This objective applies notably to high pressure core melt sequences.

Low pressure core melt sequences have to be dealt with so that the associated maximum conceivable releases would necessitate only very limited protective measures in area and in time for the public. This would be expressed by no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in consumption of food.