CHAPTER C: COMPLIANCE WITH ALARP PRINCIPLE

1. INTRODUCTION

A fundamental safety principle that is applied to UK nuclear power plant design and operation is that the radiation doses to workers and the general public shall be as low as reasonably practicable (ALARP). This principle requires that all measures are taken during design and operation of the plant to minimise radiation doses, providing the cost of such measures is not disproportionately large compared with the benefits achieved. The UK Health and Safety Executive (HSE) has proposed levels of risk from operation of nuclear power plants that would be societally acceptable in the UK [1]. They have defined a lower threshold of risk (the ‘Broadly Acceptable’ risk level), below which it should not be necessary to consider to the benefits of additional plant modifications.

This Chapter of Volume 1 of the EPR Fundamental Safety Overview document presents preliminary arguments that the processes applied to optimise the safety of the EPR design and achieve pre-licensing in France will enable an ALARP position to be demonstrated for a UK EPR, subject to further confirmatory studies to be carried out later in the pre-licensing process.

The EPR is a Generation III PWR design developed using experience gained by EDF, the German Utilities, Framatome and Siemens in the design, manufacturing, construction, and operation of PWRs (corresponding to over 1000 reactor-years of operation). The reactor has been designed to meet safety specifications developed by the French Nuclear Regulatory Agency (DGNSR) as set down in the EPR Technical Guidelines [2]. These guidelines were developed following an extensive optioneering process carried out in France and Germany between 1987 and 2006, on the design of a Generation III PWR suitable for construction in European countries. The outcome of optioneering exercise was reviewed by independent safety experts from several European Countries and the USA, on behalf of the French and German regulatory agencies. However the TGs in [2] are deterministic in nature and no requirements are stated to estimate individual risk, or to demonstrate that risks are ALARP, which mirror current UK practices.

The present Chapter presents the different steps taken over the past 20 years to optimise the EPR design in order to minimise the risks to the public and workers from EPR operation. The report describes the organisations involved in the safety design of the EPR, and the extent of independent scrutiny that was applied by European regulatory authorities.

With regard to the current operating PWR plants, it is demonstrated that EPR design contains enhanced protection measures including systems to prevent and mitigate the consequences of severe accidents, and to protect against internal and external hazards. It is considered that the risks to members of the public and workers from EPR operation are well below those from Generation II nuclear power plants currently operating, and will, in the UK context, be below the levels that would be considered Broadly Acceptable under UK principles of ALARP in [1].

2. ORGANISATIONS INVOLVED IN EPR DESIGN AND LICENSING

The three main bodies involved in the design and licensing process for the EPR have been the design organisations, the safety authorities (in France and Germany) and the bodies appointed to provide technical support to the safety authorities. A description of these bodies and their roles is given below.

Appendix I summarises the role of different organisations during the EPR basic design phase. Details are provided below.

2.1. DESIGN ORGANISATIONS

The EPR design organisations consist of reactor vendor and utility companies from France and Germany.

In 1987 the vendor companies Framatome and Siemens began cooperation to develop and commercialise a common PWR design aimed at the international export market. The aim of the collaboration was to pool the experience of the two companies in order to share the huge effort involved in developing a new reactor design. The companies founded a joint subsidiary company Nuclear Power International (NPI) to lead the work. Within a short period, French and German electrical utilities had joined the project, which rapidly replaced other reactor development work underway in France and Germany. The new reactor design was renamed EPR.

NPI led the organisation for the EPR design from 1990 up to the end of the Basic Design Optimisation phase (BDOP- see § 3 below) in 2000. Once the detailed design phase began, EDF took over as the lead design organisation for the EPR project to be built in France.

2.2. SAFETY AUTHORITIES

Following the decision 1989 to launch a Franco-German collaborative research and development program to design a third generation nuclear reactor, the French and German Nuclear Safety Authorities (ASN and BMU) created a joint safety directorate, (DFD) to oversee the project. Cooperation agreements were signed at the same time between the French and German technical support organisations (IRSN and GRS), and between the independent safety advisory groups supporting both regulators (the GPR in France and the RSK in Germany).

At that time, the French and German Safety authorities were as follows:

- French Nuclear Installations Safety Directorate (DGSNR- formerly DSIN/ASN). This body is attached to the Ministries for Industry, for Health and for Environment. Its main duty is to conduct or monitor the regulatory procedures a required under the Basic Nuclear installation Decree (December 11, 1963).

- German Federal Ministry for Environment and Reactor Safety (BMU).

2.3. BODIES PROVIDING TECHNICAL SUPPORT TO NATIONAL SAFETY AUTHORITIES

During the evaluation and assessment of the EPR design, the French nuclear safety authority (DGSNR) was supported by the following organisations:

- The Institute for Nuclear Safety and Protection (IRSN, formerly IPSN). The role of the IRSN, which consists of 1500 professional scientists and engineers, is to provide technical support to the DGSNR in relation to reactor safety and licensing and to conduct analytical studies, research and other work relating to nuclear safety on behalf of ministerial departments and interested organizations. Results of IPSN studies are submitted to GPR (see below) and/or to DGSNR.

- The Central Committee for Pressure Vessels (CCAP, article 26 of decree 99-1046 of 13 December 1999 concerning pressure vessels) is a consultative organisation. It comprises members of the various administrations concerned, persons chosen for their particular competence and representatives of the manufacturers and users of pressure vessels and of the relevant technical and professional organisations,

- For particular supervision of the more important pressure vessels in nuclear installations, it set up a Standing Nuclear Section (SPN), the role of which is to issue recommendations on application of pressure vessel regulations to the main nuclear steam supply systems.

- The Standing Group for Nuclear Reactors (GPR). This is an advisory body established by the French Minister of Industry, to support the DGSNR. It consists of 30 professional scientists and engineers from France and other European countries and the USA, and French government departments, who are specialists in the fields of safety, construction, commissioning, operation and decommissioning of nuclear reactors. The role and the membership of the GPR is given in Appendix II.

On the German side, the BMU was supported by the Gesellschaft für Anlagen und Reaktorsicherheit (GRS) which plays the same role as the IPSN (IRSN) in France, and the Reaktor Sicherheitskommission (RSK) which plays the same role as the French GPR.

After the completion of the Basic Design Phase of the EPR in 1998 the BMU withdrew from the EPR design evaluation project. However experts from Germany continued to participate in the project as invited members of the GPR.

It is important to underline that GPR members are independent of reactor vendor and utility organisations. GPR members are appointed on the basis of distinction achieved in the scientific and technical fields in which they have expertise.

3. MAIN PHASES OF THE EPR PROJECT

The phases of the EPR design process are summarised in Appendix III and described below.

In 1985 EDF, CEA and Framatome launched a research program on an evolutionary design for the next generation of PWRs, taking the N4 plant series (Chooz 1&2, Civaux 1&2) as a starting point. Studies were carried out on the design of key equipment and phenomena such as the fuel, core, safeguards systems, C&I and MMI, severe accidents, etc. Considering that the next generation of the plant was likely to be built early in the 21st century, the research program was named REP 2000.

In 1989, NPI was created and the REP 2000 project was absorbed into a wider project to design and license the next generation of NPPs in France and Germany. Contacts were established with the French and German safety authorities. Both authorities responded positively and the French safety authority specified its initial views on possible improvements that could be introduced in the design of future NPPs [3].

In 1993 NPI submitted a conceptual design specification for EPR to the French and German Safety Authorities.

In the period 1993-1995 the safety authorities, supported by ISPN/GPR and GRS/RSK, carried out a detailed assessment of the EPR design specification, and requested certain enhancements and changes. The outcome was an updated design specification which presented the detailed design requirements for the EPR plant and the rationale for the options chosen.

In 1997 a Basic Design Report was issued on the basis of the specification, which formed the first version of the EPR preliminary safety report.

Further work was carried out in the period 1997-1999 to improve and optimise the design and the investment costs within the fundamental design specification, resulting in an updated Basic Design Report (Basic Design Optimisation Phase). The BMU withdrew from the joint project during this period.

Subsequently, in the period 1999-2001, there was a further Post Basic Design Optimisation Phase, in which further design consolidation took place, particularly with regard to radiation protection of workers and environmental emissions.

In 2000 the French Safety Authority DGNSR/IRSN issued Technical Guidelines [2] which set down design principles for future NPPs that would be acceptable for construction in France. These TGs set down the initiating events that must be addressed within the design basis for new NPPs and defined certain multiple failure sequences and core melt sequences that must be considered in the design. The TGs also included requirements for design against external hazards. The functional requirements of safety systems and principles for their safety classification were stated, including assumptions that must be made with regard to single failures, and the equipment unavailability due to maintenance.

Following issue of the TGs, further enhancements were made to the EPR design, and three further updates were produced to the EPR safety report.

In Finland, the application file for the Construction Licence of the first EPR, Olkiluoto 3, was submitted by the utility TVO early January 2004, the Finnish Safety Authority STUK issued its statement and safety evaluation on January 21, 2005, and following the Construction Licence was awarded on February 2005.

In 2005 EDF decided to apply for a license to construct the first French EPR at the Flamanville site in Normandy. A pre-construction safety report for the FA3 EPR unit was approved by the DGNSR in 2006. The public version of this document has been translated into English and a shortened version, with specific references to the FA3 site removed, is included as Volume 2 of this pre-licensing submission.

Since 1993, some 6 EPR Safety Reports and approximately 180 supporting technical references have been produced and assessed in detail by independent experts in IRSN/GPR (and by GRS/RSK up to 1998). During this period approximately 80 evaluation reports were issued by IRSN/GRS, some 200 meetings were held between the EPR project and the regulatory agencies, and close to one million hours of work were carried out within the EDF-SA engineering functions alone.

It is thus seen that the EPR design proposed for construction in the UK has undergone a 20 year process of design optimization to maximise the safety of the plant within the constraints of practical constructability. This process has a large similarity with the process of optimization of design to achieve an ALARP position envisaged in [1]. The EPR design process has been carried out in consultation with regulatory authorities in France (and Germany initially), and all design documentation has been independently reviewed and scrutinised by a large body of experts in organisations which were independent of vendor and utility companies.

3.1. GLOBAL ASSESSMENT: THE "TECHNICAL GUIDELINES"

French and German Safety Authorities decided in 1995 that "guidelines for future PWRs" should be developed with oversight provided by the GPR & RSK standing commissions. IPSN and GRS submitted a proposal for the structure and contents for the development of these guidelines which was adopted. The outcome of this work was the publication of the "Technical Guidelines" (TGs) which brought together all recommendations of GPR and RSK (or German experts) during the period 1993 - 2000. The TGs were developed from the documents "proposal for a common safety approach for future PWRs" issued in 1993 (see Appendix IV A) and "development of a common safety approach on five key issues important for early resolution in the design process" issued in 1994 (see Appendix IV B).

The Technical Guidelines implement and refine the EPR conceptual safety design features identified in 1995. Several steps were involved in their production, as follows:

- The first step was to propose a structure allowing the GPR/RSK recommendations to be implemented in such a way that the technical guidelines can be used in both French and German regulatory processes. This was the objective of the work performed in 1997,

- The second step was to organise the GPR/RSK recommendations in the adopted structure. This work was performed in the first part semester of 1998,

- In the middle of 1998, IPSN/GRS began to turn the text of the GPR/RSK recommendations into guideline text, rearranging and making some changes in the material (without changing the substances of the original GPR/RSK recommendations). An intermediate version of the Technical Guidelines was issued by IPSN/GRS at the end of 1998. This document was reviewed by GPR/RSK members and by the EPR design organisation, leading to the issue of an updated version in 1999.

The final text of the Technical Guidelines was adopted by GPR and German experts during the plenary meetings held on October 19 and 26, 2000. The final contents list is presented in Appendix IV C.

3.2. REVIEW AND VALIDATION OF EPR DESIGN OPTIONS BY EDF

EDF carried out a design review in the early preparatory phase of the EPR project, and later on during the detailed design phases after the issue of the Technical Guidelines (see Appendix III). In the period in between, NPI had primary responsibility for the EPR technical design review. Work performed during the NPI period is described in § 3.3.

The phase of the project after the Technical Guidelines were issued is of the greatest interest for the purpose of the present report, as the French Safety Authority, although being positive overall, asked for improvements in some features of the EPR design.

Several letters were sent by EDF to the French Regulator in late 1999 and early 2000, making commitments to improve the design in areas such as:

- o demonstration that some accident sequences (reactivity accidents, containment bypass sequences) could be “practically eliminated” from the design basis,
- o reducing the consequences of accidents involving core melt (hydrogen mitigation, cooling of the corium, ..) and the design of the systems used in these situations,
- o role of the containment building and the peripheral buildings in the design of the “containment function”,
- o definition of reference events considered in the design basis of the EPR, in particular with regard to cooling of the fuel inside the spent fuel pool,
- o consideration of earthquakes in the design and the combination of loads to be assumed in design conditions with loads due to the basis design earthquake,
- o the program to be implemented for the equipment qualification, to ensure demonstration of qualification before start-up of the first EPR plant,
- o design objectives for radiation protection, radioactivity releases and radioactive waste production,
- o consideration, at the design phase, of the specific needs and provisions to facilitate decommissioning of the EPR.

In addition, and on the basis of these commitments, design reviews on specific items were also conducted at EDF’s initiative, such as on the severe accident mitigation strategy, the strategy for radiation protection etc.

3.3. REVIEW AND VALIDATION OF EPR DESIGN OPTIONS BY NPI

During the consolidation phase and the Basic Design phase (see Appendix III), the work performed in accordance with the Basic Design Contract between EDF, German Utilities, Framatome, Siemens and NPI was organized as shown in Appendix I.2.

The following definition will help in understanding the work organization for the definition and validation of the main options considered during the Basic Design Phase.

- Contractor refers to the group of companies formed by NPI, Framatome and Siemens, these companies being jointly and severally liable to the Utilities, under the leadership of NPI.
- Utilities refer to the group of companies comprising EDF and the German Utilities.
- Designer is any of the three companies forming the Contractor (or EDF-CNEN, to the extent it is entrusted by the Contractor to perform services).
- EPR Project Leader Committee (PLC): refers to a joint group formed between the Utilities and the Contractor as project organization. It is responsible for the realization of the decisions taken by the ESC and the EPD with regard to technical questions. Furthermore, the PLC is responsible for the execution of the work in due time and appropriate manner according to the instructions of the EPD.
- EPR Project Directorate (EPD): refers to a joint group formed between the Utilities and the Contractor as project organization. It makes decisions on all technical questions, apart from those reserved for the EPR Steering Committee (ESC) or submitted by the EPD to the ESC for a decision.

The Basic Design work was performed within the orientations and main technical features defined during the Conceptual Design Phase and the Consolidation Phase. These orientations and technical features were described in a set of sheets which constituted the Main Feature Files (MFF). The Basic Design activities allowed validating and complementing main features which were by definition general and preliminary. In some cases nevertheless departures from this MFF had to be considered.

The EPR Project Directorate was in charge of deciding such changes upon proposal of the Project Leader Committee, or when those changes appeared significant enough to report for approval to the EPR Steering Committee.

3.3.1. Work of the Committees and Technical Working Groups

The following committees and working groups met on a regular basis to make decisions and carry out actions within their respective field of responsibility.

3.3.1.1. EPR Steering Committee (ESC)

The EPR Steering Committee was entitled to take any decision concerning the Basic Design Contract or to grant the right to take decisions to another project body.

3.3.1.2. EPR Project Directorate (EPD)

The EPR Project Directorate decided on all technical questions, apart from those reserved for the ESC or submitted by the EPR Project Directorate to the ESC for decision.

68 EPD Meetings were held: 27 during Conceptual Design Phase and Consolidation Phase, and 41 during Basic Design Phase and Basic Design Optimization Phase.

3.3.1.3. EPR Project Leader Committee (PLC)

The EPR Project Leader Committee was responsible for the realization of decisions taken by the ESC and the EPD concerning technical questions. Furthermore the PLC was responsible for the execution of the work in due time and appropriate manner according to the instructions of the EPD.

The meetings of the PLC were organized in order to review the progress of the project (Progress Review Meeting), in order to control the activities of the working groups and to prepare the results for discussion in the EPD.

92 PLC Meetings were held: 36 during Conceptual Design Phase and Consolidation Phase and 56 during Basic Design Phase and Basic Design Optimization Phase.

3.3.1.4. Working Group Meetings

Subject to approval of the EPD, the PLC established working groups for special problems. The working groups generally consisted of representatives of EDF and the German Utilities, as well as of members of the Contractor organizations and EDF-CNEN when acting on behalf of the Contractor.

The meetings were organized by the project leaders who nominated qualified participants from their organization. The following items were considered by permanent working groups with regular meetings:

1. Safety principles
2. Functional engineering and transient analysis
3. Systems and process
4. Reactor Core
5. Containment
6. Layout, Civil (and. radiation protection)
7. I&C
8. Electrical systems
9. Components/Equipment units of the primary circuit
10. Severe accidents and. radiology

The working groups were involved from the very beginning in such a way that they were aware of ongoing engineering work and of the intermediate results, so they would have a common understanding of the design choices.

3.3.2. Technical Reviews

In addition to the periodic meetings, Technical Reviews were organized with the participation of the Contractor and the Utilities. Reviews were performed on specific subjects, such as:

- safety and systems, in particular severe accident reviews and systems reviews,
- layout, including building layout, accident prevention for personnel, radiation protection for personnel, fire protection, and other hazards.

A Technical Review Group was composed of experts from the Contractor and from Utilities and the Project Leader Committee. At the meetings, experts from the Contractor and Utilities presented their assessment reports, under the coordination of NPI, on the technical issues addressed by the Technical Review. The members of the Technical Review Group could request any further explanations and justifications. Their conclusions were presented in a report to the Project Leader Committee (PLC).

NPI were responsible for documenting the report of the Technical Review Group and distributing it to the PLC, the PED and the participants to the Technical Review. After approval by the PLC, the report became effective. NPI then issued a complete report of the meeting, including the comments of the various organizations represented at the meeting and rationales for the conclusions reached.

Technical reviews were held during the course of the Basic Design of the EPR, on different topics such as ECC Mode, Severe Accidents, I&C, Systems, Layout:

- ECC Mode – September 12, 1995
- Severe Accidents – December 1, 1995
- Severe Accidents – April 11, 1996
- I&C – September 17, 1996
- Layout – March 19, 1996
- Systems – February 20, 1996
- Systems – March 5, 1996
- Systems, December 16, 1997

Following the completion of the Basic Design there were further design phases, including the Design Optimisation and Detailed Design phases (see Appendix III). Technical Review Group meetings continued during these phases, addressing topics such as the design of the containment building and its liner, design of ventilation systems, selection of steam generator tube material, human factors, human-machine interface, radiological protection and environmental impact.

4. OUTCOME OF DESIGN PROCESS

4.1. GENERAL DESIGN

International studies on the safety approach for the next generation of NPPs (for example, those conducted by the IAEA) have stressed the importance of strengthening the defence in depth concept, and achieving greater independence between the barriers protecting against releases of radioactivity to the environment. These measures require, amongst other things, explicit consideration of severe accidents in the design of the containment system.

The main elements of the defence-in-depth concept that have been used as a basis for the design of the EPR are:

- Balance between the different levels of protection,

- Degree of independence of these different levels of protection,
- Achieving an appropriate balance between the prevention and the mitigation of accidental situations,
- Achieving an adequate level of conservatism in the design and in the protection of the barriers,
- Increased emphasis on shutdown states and on specific events that may result in potential containment bypass.

The above considerations have led to a number of design enhancements for EPR compared with previous Generation II NPPs. Some of the significant enhancements are reviewed below.

4.2. EPR DESIGN REVIEW RESULTS

4.2.1. Containment design

To meet the safety objectives for EPR, modifications to the containment design were required to improve protection against uncontrolled environmental releases in all design conditions considered, including core melt sequences and external hazard conditions. In such accident situations, the requirement was that the structural integrity of the containment and its leak-tightness must be preserved.

The initial concept chosen for EPR containment building was a double-wall containment based on technology derived from the current French N4 containment design. The inner wall is a prestressed concrete shell. The outer wall is made of reinforced concrete and is part of a global “aircraft shell” that protects most parts of the nuclear island from external hazards. The two walls are separated by an inter-wall annulus, which is maintained at a negative pressure so that leakages from the inner containment can be collected and filtered before being discharged to atmosphere via the stack. The double-wall containment building stands on a reinforced concrete basemat.

Design studies for the EPR containment started on the basis of the options described in the Conceptual Safety Features Review File [3]. The main technological and design principles chosen to satisfy the required specifications and their evolution are as described below:

- An internal containment volume of 75000 m^3 was initially chosen to reduce the risk of a global hydrogen detonation. This resulted in the maximum concentration of hydrogen in dry air within the range 10 and 13 %. Taking into account the 241 fuel assemblies in the core and a target maximum hydrogen concentration of 10 %, optimisation studies resulted in the free volume of being increased to 80000 m^3. With this change, the maximum containment pressure was limited to 4,5 bar (absolute) in the event of a LOCA, increasing to 6 bar in the case of a hydrogen deflagration,

- The containment outer wall, made of reinforced concrete, is designed to withstand loads resulting from external hazards (earthquake, explosion, airplane crash, etc) and leak for recovery. Compared to the loads used in the design of the French N4 plants, the loading combinations due to external hazards considered for EPR are greatly increased. In particular the seismic level considered corresponds to a peak ground acceleration of 0.25 g and the external explosion load corresponds to an incident pressure wave of 100 mbar. The design of EPR to withstand airplane crashes initially considered only loads due to the impact of military jet fighters. Two curves (C1 and C2) have been issued to consider the effects of the induced vibrations and the local penetration of missiles. However, after the 9/11 events, the design loads on the outer wall were considerably increased to embrace a wide range of potential terrorist attacks, including the deliberate crashing of a large commercial airplane,

- To ensure high containment leaktightness, specific design criteria have been implemented to prevent direct leakages from the Reactor Building to the environment. These include a requirement that all pipes carrying radioactive substances which pass through the containment wall pass into sealed peripheral buildings so there is no direct release route from inside the containment to the environment. Therefore the peripheral buildings are considered to form part of the containment function,

- In addition to this initial design requirement, a complete study of all potential bypasses of the containment function due to the failure of equipment connected to pipework passing from the containment building to peripheral buildings, has been performed. This study has resulted in changes to the design of some fluid circuits, including use of complementary isolation methods (see below),

- In the initial design, the leak tightness of the containment was achieved using a partial composite liner on the inner face of the internal wall. Tests were conducted on a large scale mock-up (with and without the composite liner) to gain experience of the behaviour of such a system, using dry air and air/steam mixtures in pressurized conditions. On the basis of the results of these tests, and experience feedback from current French plants, it was decided to change to using a steel liner inside the containment in order to improve its leak tightness, particularly the case of severe accidents conditions.

A reference document has been issued during the detailed design phase summarising all the safety requirements developed during the design of the EPR containment and the EPR civil work structures. It is presented in Volume 2 subchapter C.3 of this UK EPR pre-licensing submission.

4.2.2. Design improvements to avoid containment bypass events

Consideration has been given to fault sequences in which radioactive materials might be released to the environment via fluid systems connected to the Reactor Coolant Circuit (e.g. the Safety Injection/Residual Heat removal system, the Chemical and Volume Control System, the Extra Boration System etc.), effectively bypassing the containment function.

The risk of containment bypass via Steam Generator Tube Rupture (STGR) has also been analysed in the containment design. Design provisions adopted for EPR, particularly the requirement for the maximum head developed by the safety injection pumps to be below the set pressure of the safety relief valves on the secondary system, plus the requirement for automatic shutdown of Chemical and Volume Control system charging pumps on detection of high water level in the Steam Generators, reduce the risk of primary coolant bypassing the containment through the secondary system.

Functional analysis and quantification of the potential initiators to evaluate the risk of bypass for each of the scenarios considered using PSA methods suggest that the risk of bypass is very low (assessed at few 10^{-9}/yr for core melt accidents).

4.2.3. Design for mitigation of severe accidents

As stated in the general safety objectives for the next generation of NPPs, a significant reduction of potential radioactive releases due to all conceivable accidents has to be achieved compared to the current plants, including for core melt accidents. This means that:

- Accident situations involving core melt which would lead to large early releases have to be “practically eliminated”. Such an objective applies in particular to high pressure core melt sequences,
- Low pressure core melt sequences have to be dealt with so that the maximum conceivable releases would necessitate only very limited public protective measures in the vicinity of the plant (only limited sheltering).

Due to these requirements, and considering on the basis on the defence-in-depth concept that the core melt has to be cooled outside the reactor pressure vessel, specific means have been designed and implemented to protect the containment building, and in particular its basemat from core melt situations.

For the basemat protection, the original solution selected for the EPR was to allow spreading of the core melt debris over a large area outside the reactor cavity. The overall aim of the concept was to stabilize the melt by water cooling within the containment in order to avoid a long-term source term for fission products, due to intolerable thermal loads or damage to structures which could result in loss of the containment function.

The spreading area which is maintained in dry conditions during normal operation is designed to avoid the hydrogen production due to the interaction between the basemat concrete and the corium. It is connected to the in-containment water storage tank (IRWST) by pipes which flood the corium once it has been spread in the compartment.

In addition, and in order to limit the pressure increase and to decrease this pressure inside the containment as rapidly as possible, a Containment Heat Removal System has been implemented to cool the water used to fill the spreading compartment. This system, which has no active components inside containment, is a two train system cooled by an intermediate active system with power supplied by two specific diesel generators.

With regard to control of hydrogen concentration, several solutions have been considered including recombiners and igniters to prevent global detonation or deflagration due to high H2 concentrations.

The design measures for mitigating the consequences of low pressure core melt scenarios have evolved significantly during different phases of the EPR design process. Numerous studies and tests have been performed which have led to improvements in the design of specific elements or components such as:

- The capacity of the reactor cavity to collect all the corium produced that could be produced inside the vessel due to the fuel melting using a “iron-oxide” sacrificial concrete in the cavity pit,
- The positioning and the design of the gate which contains the corium before discharging it into the spreading area through the melt discharge channel,
- The simplification of the design of the components protecting the basemat by eliminating the initial zirconia layer, replacing it by a sacrificial concrete with an appropriate thickness and extracting the heat flux from the corium using cooling channels implemented inside the basemat underneath the spreading area.

4.2.4. Systems design: Passive features investigated for the EPR

The idea of performing safety functions by passive means is not new. All existing PWRs employ passive features like accumulators, gravity-driven control rod insertion or natural circulation in the primary circuit. Besides these, additional passive features have been included in the EPR design such as:

- larger SG and pressurizer volumes providing increased thermal inertia, thus slowing plant response to upset conditions,
- initial SIS valve line-up (suction from IRWST) meets long term cooling needs without realignment,
- lower core elevation relative to the cold leg cross-over piping which limits core uncovery during small break LOCAs,
- absence of lower head penetrations on the RPV for in-core instrumentation, thus eliminating one potential failure mechanism and failure location,
- passive pressurizer safety valves for both overpressure protection and prevention of spurious opening (passive opening under pressure increase, passive closing under pressure decrease),
- a large dedicated spreading area outside the reactor cavity to prevent the molten core-concrete interaction, by spreading and subsequent flooding of the corium,
- a large water source in the IRWST located inside the reactor building, draining by gravity into the reactor cavity and the corium spreading area,
- a double wall containment with a reinforced concrete outer wall and a pre-stressed concrete inner wall and an intermediate space maintained passively under a small sub-atmospheric pressure.

In addition to the above features, about twenty passive features were evaluated at the beginning of the conceptual phase of the EPR. The depth of evaluation of specific features depended on interest for their application. About half were briefly examined and dropped without further evaluation; the others were assessed in more detail. The principal passive features which were investigated for possible implementation in the EPR, but not retained in the final design, are described in appendix V.

4.3. CONTRIBUTION OF PSA STUDIES IN THE EPR DESIGN

A general objective for the design of the next generation of nuclear power plants has been to reinforce the main elements providing defence in depth of the plants. To achieve this objective, the design has been made on deterministic bases supplemented by the use of probabilistic methods. Reactor operating experience and in–depth studies such as PSA studies as well as improvements in knowledge of physical phenomena during accident situations like core melt were been taken into account.

PSA can be used to quantitatively demonstrate implementation of the defence-in-depth concept as well as to show that a balance has been achieved between levels of protection and that the levels are independent of one another.

Different PSA studies have been performed at the design stage of the EPR to support the choice of design options, including redundancy and diversity of the safety systems. PSA has also been used to select or reject changes to the main EPR options during the Basic Design Optimisation Phase.

As an example with regard to the availability and reliability of the emergency electrical power supply, PSA was used to determine the number of diesel generators (four main diesel generators, of which two are backed up by two smaller diesels of diverse design) and to demonstrate the sufficient independence and diversity exists amongst the two types of diesel generators.

PSA studies are also used to reduce the impact of common cause failure between redundant systems. This is the case of the Containment Heat Removal System for which several designs of its cooling chain have been compared, leading to the final design that includes a dedicated cooling chain, independent of the Component Cooling Water System.

In addition to supporting the choice of design options, PSA has been used to validate the list of events considered in the EPR design basis and to define the list of sequences addressing multiple failures to be considered in the framework of the prevention of core melt scenarios (RRC-A).

The systematic use of PSA to develop an optimised design which minimises risks, subject to practical considerations of cost and constructability, is considered analogous to the ALARP approach to design conventionally applied in the UK.

4.4. SUMMARY OF DESIGN MEASURES RELATIVE TO ALARP REQUIREMENTS

It has been shown that the EPR design process has resulted in addition of enhanced protection to prevent and mitigate the consequences of severe accidents, to protect against external hazards, to improve the capability of the containment function, including reducing the likelihood of radiological releases due to containment bypass sequences, and to increase the use of passive safety measures. The safety improvements have been backed by use of PSA to develop an optimised design which minimises risk.

The EPR design has therefore undergone a 20 year process of design optimization to maximise the safety of the plant within the constraints of practical constructability. This process is closely analogous to the process of optimization of design to achieve an ALARP position envisaged in [1].

Preliminary PSA results presented in Chapter R of Volume 2 of this UK EPR Fundamental Safety Overview, show that the probability of core melt in an EPR due to plant faults will be $<10^{-6}$/yr, and that the probability of a large radiological release (i.e. core melt + loss of containment function) is below 10^{-7}/yr. Chapter R also contains a preliminary assessment of the risk of core melt due to hazards, and concludes that this is likely to be comparable to that due to plant faults. Thus it may be concluded that the frequency of a large radiological release from EPR operation will be or order 10^{-7}/yr. At this level, the risk to members of the general pubic off-site from large radiological releases would be in the Broadly Acceptable region defined in [1].

The risk to individuals due to smaller releases, arising from containment bypass sequences has not been calculated in the current EPR PSA, but such calculations are to be undertaken for a UK sited EPR (see Section 5.2 below). However the measures taken to reduce the likelihood of containment bypass events (see 4.2.2 above), give confidence that these risks are also likely to be small, and within the ‘Broadly Acceptable region. Therefore there is confidence that the current plant design will meet the principle of ALARP in purely probabilistic terms. This conclusion will be confirmed by further PSA studies in progress.

5. OPTIMISATION OF DOSE TO WORKERS AND THE PUBLIC IN NORMAL OPERATION

To establish an ALARP position, under UK ALARP principles, the dose to workers and members of the public from plant operation, must also be as low as reasonably practicable. These aspects are discussed briefly below.

5.1. DOSE TO WORKERS

The EPR design philosophy involved extensive design optimisation to minimise worker dose. The objective was to apply an optimisation approach to radiological protection similar to that applied to safety. This has been accomplished by improving the reactor design in relation to the best NPP units currently operating in France and Germany with regard to worker collective dose. Details of the design optimisation are provided in Chapter L.4 of Volume 2 of this Fundamental Safety Overview, and a summary is given below.

The EPR approach to dose optimisation was to improve the design to both reduce the source term associated with plant operation and maintenance, and to reduce the amount of exposed work.

The EPR design source term has been reduced by the following design changes:

- optimising the use of stellite in the reactor vessel internals and valves,
- modifying the pressuriser layout to reduce the dose from maintenance activities in the safety valves area, and from inspection of the pressuriser heaters,
- inclusion in the reactor building of an area dedicated to storing the pressure vessel head (with appropriate shielding),
- removal of "hot spots" by eliminating pipe connections using socket welds on pipework carrying radioactive fluids, by chemistry optimisation, and by reducing the amount of antimony and chromium in primary system components.

The amount of exposed work (subject to radiation) has been reduced by the following design choices:

- design changes to allow maintenance operations to be carried out in the reactor building during power operation, immediately before and immediately after outages. Provision of improved shielding and ventilation in the reactor building zone to which at-power access is permitted ensures a net reduction in the collective radiation doses associated with outages,
- increased use of bolted connections on key equipment (pressuriser heaters, control rod drive mechanisms etc),
- increased size of primary and secondary manways,
- modification to the steam generator waterbox layout to give easier access to peripheral tubes,
- reactor vessel head heat insulation removable as a single unit,
- absence of a forced ventilation device for the Control Rod Drive Mechanisms (CRDMs) resulting in the elimination of opening and closing operations for the CDRM ventilation system air duct,
- improvement to reactor vessel level instrumentation to reduce required maintenance operations,
- routing of the ex-core instrumentation into the reactor-vessel pit through the pool concrete wall to reduce operations associated with instrumentation system covers at the bottom of the pool,
- optimisation of the fuel handling operation duration,
- installation of improved shielding around active equipment,
- use of modular maintenance valves.

In parallel with the dose assessment of the design developments, optimisation studies were carried out for the high dose activities by the designers responsible for installation, materials and operation.

Taking into account proven modifications and radiological protection modifications under consideration for EPR, the value of the optimised dose is estimated as 0.366 man.Sv per year per unit (achievable over a 10 year operating period). Further optimisation work is in progress to reduce the dose further to a target level of 0.35 man.Sv per year per unit. This is a significant improvement on the collective dose for the best operating unit of the French fleet, GOLFECH 2, which has achieved 0.44 man.Sv per year over a full cycle of ten years. It is also only a small fraction of the design dose level for earlier PWR plants such as the UK Sizewell B PWR.

These measures give confidence that the operator dose due to plant operation has been reduced to minimum practicable level by optimisation of the plant design, and therefore that ALARP principles have been met in respect of worker risk from normal operation of the plant.

5.2. DOSE TO MEMBERS OF THE PUBLIC DUE TO NORMAL OPERATION

Achievement of an ALARP situation requires that radiation doses to the general public due to normal plant operation are minimised and are as low as reasonably practicable.

The EPR design has been optimised to reduce off-site doses to a minimum, and to achieve significant improvements compared with existing French and German NPPs.

To achieve a realistic assessment of radioactive release values for the EPR, a pragmatic approach has been adopted, consisting of comparing the EPR design with those of existing units and considering the decisive components of the production/treatment/release chain. This comparative approach comprises the following steps:

- comparison of primary source terms,
- identification of main release paths, and quantification for existing plant fleet (step strongly relying on analysis of experience feedback),
- comparison of release path designs for EPR and existing plant fleet,
- assessment of influence of EPR design changes on estimated release values for each path.

Further details are given in as described in Chapter G of the Volume 1 Head Document.

6. FUTURE STUDIES

6.1. FUTURE FA3 STUDIES TO BE CARRIED OUT BEFORE CORE LOADING

The construction license (DAC) for the FA3 EPR was issued by the French Safety Authority in the spring of 2007. Assessment by the French Safety Authority will continue up to the granting of the operation license. On the basis of current instructions and commitments in the Preliminary Safety Report, a work programme has been agreed between EDF and the French Safety Authority and its technical support (IRSN) that will be required to be completed before core loading. The agreed work is as follows:

- Civil work : verification of the implementation of the requirements in the detailed design of the buildings of the nuclear island, conventional island and the pumping station,
- Main mechanical components : organisation and control of the manufacturing of the main mechanical components in accordance with their design specifications,
- Equipment qualification with regard to accident conditions, including severe accidents situations : validation of the parameters (pressure, temperature, irradiation, duration, etc.) to be considered for each family of equipment to be qualified,
- Refinement to the safety approach : final adjustments to some elements of the safety approach such as the component classification, the methodology to assess some hazards, etc.,
- Systems design : completion of a detailed design review for certain systems such as the Safety Injection System, Containment Heat Removal System, Fuel Pool Cooling System, etc.,
- Accident studies : verification of the result of previous fault studies including RRC-A and RRC-B sequences,
- I&C and MMI : final assessment of the I&C architecture and the associated systems,
- Radiation protection, gaseous and liquid releases, wastes : final estimation and validation of the quantities produced during the normal operation of the plant,
- PSA : Extension of PSA studies to include level 1 PSA for the fuel building, PSA dedicated to hazards and completion of level 2 PSA,
- Radiological consequences : validation and implementation of the methodology to be used for the calculation of the radiological consequences of severe accidents,
- General Operating Rules: validation and verification of methods for creating and issuing Emergency Operating Procedures, Equipment Surveillance Procedures and the Limiting Conditions for Operation.

In all cases, the depth of assessment to be performed in the next four years will depend on the level of assessment realized before issue of the construction license.

6.2. FUTURE ALARP STUDIES TO BE CARRIED OUT FOR THE UK EPR

The EPR design optimisation process will be summarised in the Pre-Construction Safety Report (PCSR) for the UK EPR which will be prepared later in pre-licensing. This document will confirm that the design meets UK ALARP principles in respect of the risk to members of the public offsite and to workers.

The PCSR will extend the existing EPR PSA model to carry out a preliminary Level2/Level 3 PSA for a UK sited EPR, which will be bounding with respect to possible UK sites. The PSA will enable the frequency of radiological releases of different magnitudes to be estimated, along with the corresponding radiological risk to workers and vulnerable individuals off-site. The study will cover potential initiating events affecting the reactor and the fuel storage pool, during all phases of plant operation, and will include major uncontrolled releases of radioactivity involving fuel melt and smaller releases due to containment bypass sequences. It is expected that this study will confirm that the risk to members of the public and workers due to the EPR will be within the Broadly Acceptable region defined in UK ALARP methodology, confirming that the plant design does not require significant further improvement under ALARP principles.

7. SUMMARY OF ALARP ARGUMENT

Although UK principles of ALARP were not applied as an integral part of the EPR design process, it has been shown that the process of design optimisation to minimise risk due to accidents and to optimise operator dose in normal plant operation are closely analogous to the formal UK approach of ALARP. In particular:

- the design has been developed by a comprehensive and systematic process to reduce risk to workers and the public,
- extensive design optioneering has been carried out in consultation with French and German safety authorities and international experts to optimise plant safety,
- both public and worker risk has been addressed within the safety design approach,
- risks due to normal operation and accidents have been considered,
- PSA methods have been used to show that the risk of a significant environmental radiological release due to accidents is at a very low level, which corresponds to the Broadly Acceptable risk level at which further design improvements would not normally be required under UK ALARP principles. Further developments of the PSA are in progress,
- design decisions and their rationale have been documented.

8. CONCLUSION

The EPR is a Generation III reactor design which has been developed to address issues of safety, public perception or risk, and economics, on the basis on the operating and engineering experience in both France and Germany. This report has described the design and licensing process applied to EPR.

The EPR design contains enhanced protection measures to prevent and mitigate the consequences of severe accidents, to protect against internal and external hazards, to reduce the likelihood of radiological releases due to containment bypass sequences, and to increase the use of passive safety measures in the design.

The safety improvements have been underwritten by PSA studies in order to produce an optimised design which minimises risks, within the constraints of practical constructability. The design process is considered closely analogous to the UK process of optimising the design of a nuclear plant to ensure that risks from its operation are ALARP. In parallel an exercise has been carried out to optimise the design in respect of the dose received by works during operation and maintenance of the plant.

As a result of the design optimisation, it is considered that the risks to members of the public and workers from EPR operation are below those from Generation II nuclear power plants currently operating, and are also below the levels that would be considered Broadly Acceptable under UK principles of ALARP.

The EPR design optimisation process will be summarised in a Pre-Construction Safety Report (PCSR) which will be prepared later in UK EPR pre-licensing process. This document will confirm that the design meets UK ALARP principles in respect of the risk to members of the public and workers. It will also present results of new Level 2/Level 3 PSA study on a UK EPR to confirm that the Broadly Acceptable levels of risk are achieved by the proposed plant design, and to confirm that significant further design improvements are not warranted under ALARP principles.

9. APPENDICES

Appendix I: EPR Organisation during the Basic Design Phase

Appendix II: Role and composition of the Standing Group for Nuclear Reactor (GPR)

Appendix III: Different phases of the EPR design

Appendix IV A: Content of the proposal adopted on May 25, 1993:"GPR/RSK proposal for a common Safety Approach for Future PWRs"

Appendix IV B: "Key issues of conceptual safety features" selected in 1994 for early decisions

Appendix IV C: Content of the "Technical Guidelines for the design and construction of the next generation of PWRs plants" adopted on October 26, 2000:

Appendix V: the EPR approach for active and passive systems

10. REFERENCES

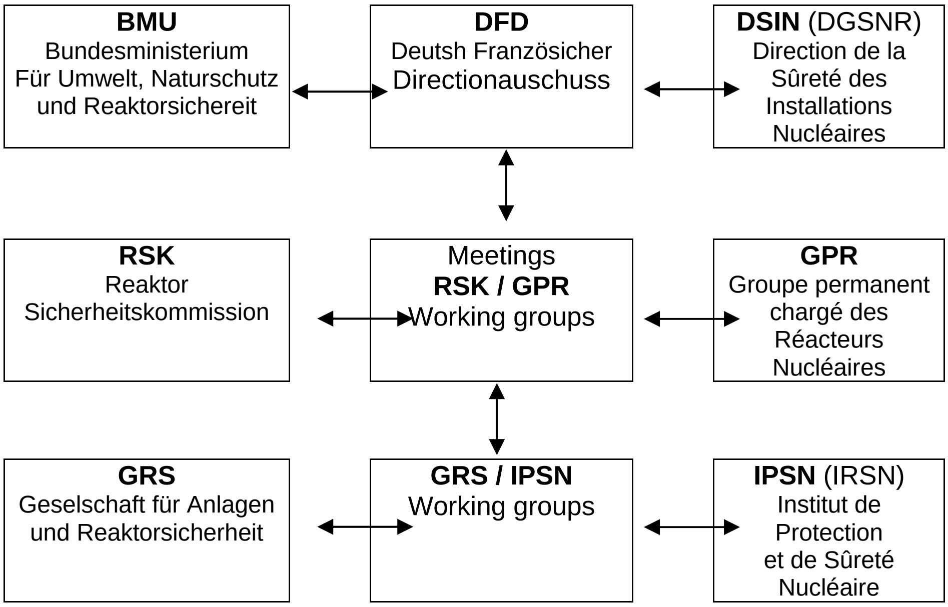
[1] Health and Safety Executive (HSE). The Tolerability of Risk from Nuclear Power Stations (ISBN 0118863681), HMSO 1992

[2] Technical Guidelines for the Design and Construction of the Next Generation of Nuclear Power Plants with PWRs, GPR/German experts plenary meetings October 19th and 26th, 2000.

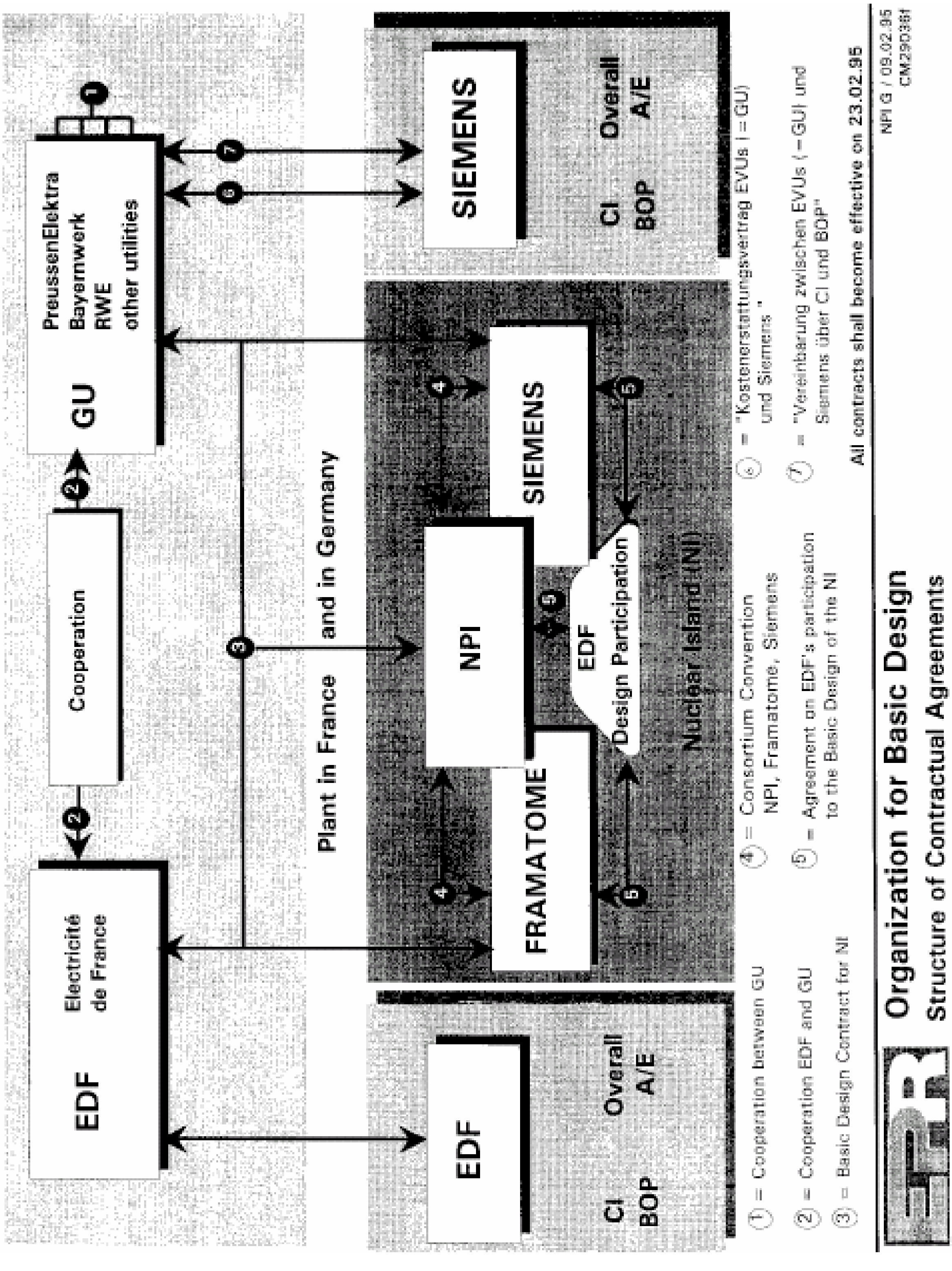
[3] EPR Conceptual Safety Features Review File; report jointly prepared by EDF, German Utilities, NPI, Siemens, Framatome; August 2nd 1993,

Appendix I : EPR Organisation during the Basic Design Phase

1 - Organisation of the relationship between French and German safety bodies



2 – Organisation of the relationship between French and German co-designers



Appendix II:

Role and composition of the Standing Group for Nuclear Reactors (GPR)

The Standing Group for Nuclear Reactors (GPR) was instituted by the French Safety Authority (DGSNR) by its decision of March 27, 1973. It is charged with the study of technical problems arising in the safety design of nuclear reactors in relation to start-up, operation, final shut-down and dismantling.

The composition of the GPR is regularly re-examined: the last decision relating to its composition was made by the General Director of DGSNR in December 2004 and indicates that the Standing Group Reactor should include:

- A president and a vice-president,
- A representative of the Council General of the Mines,
- Five representatives of the DGSNR,
- A representative of the General Directorate of energy and raw materials,
- A representative of the Management of the prevention of pollution and risks,
- A representative of the General Directorate of Health,
- Four titular experts and four temporary experts named on proposal of the Institute of Protection against Radiation and Nuclear Safety (IRSN),
- Four titular experts and four temporary experts named on proposal for an Electricité de France,
- Two titular experts and two temporary experts named on proposal of the Commissariat à l' Énergie Atomique,
- Twenty and one experts including international experts, chosen because of their particular competence.

The international expert members include representatives from overseas safety authorities including NII in the UK, STUK in Finland, GRS in Germany, SKI in Sweden and AVN in Belgium.

The Standing Group Reactor meets approximately 10 times each year on subjects concerning PWR and 3 to 4 times on subjects relating to experimental reactors.

Appendix III : Different phases of the EPR design in France

Dates	Design Phases	EPR Project deliverables	Safety bodies deliverables	Comments
1986 1991	Preparatory phase	REP 2000 safety specifications	Letter DSIN 984/91 Letter GPR 93/18	NPI founded in 1989
08/1993	Consolidation phase	Conceptual Safety Features Review File (CSFRF)	**Letter DSIN 1321/93 (see Appendix IV - A)**	
		Technical Reports	Letters GPR 94/20 ; GPR 94/28; GPR 94/42 ; GPR 94/50; GPR 94/62 ; GPR 95/04	
01/1995	Basic Design phase	Main Features Files (MFF)	**Letter DSIN 51/95 (see Appendix IV - B)**	Signature of the BD contract
		Technical Reports	Letters GPR 95/54, GPR 95/59 ; GPR 96/24 ; GPR 96/25 ; GPR 96/47 ; GPR 97/01 ; GPR 97/25 ; GPR 97/41; GPR 97/60	
10/1997	Basic Design Optimisation phase	Basic Design Report (BDR 97)		Signature of the BDOP contract
		Technical Reports	Letters GPR 98/07 ; GPR 98/15 ; GPR 98/31	
11/1998		Basic Design Report (BDR 98)		Withdrawal of BMU/RSK

02/1999		Basic Design Report (BDR 99)		
	<u>Post Basic Design Optimisation phase</u>	Technical Reports	Letters GPR 99/06 ; GPR 99/32 ; GPR 99/43 ; GPR 99/55 ; GPR 00/16	
10/2000		Letter EDF ref SN/99-1638	**Technical Guidelines (see Appendix IV - C**)	EDF leader of the design
		Technical Reports	Letters GPR 02/24 ; GPR 03/25 ; GPR 04/12	
12/2003	<u>Detailed Design Studies phase</u>	PSAR (project version) 2003	**Letter DGSNR 729/2004 EPR Safety Options**	
		PSAR (project version) 2005	Letters GPR 05/23 ; GPR 05/35 ;	Decision to launch FA3 project
05/2006	Request for construction license (DAC)	PSAR (project version) 2006	Letters GPR 06/06 ; GPR 06/21 ;	

Appendix IV A : Content of the proposal adopted on May 25,1993:
"GPR/RSK proposal for a common Safety Approach for Future PWRs"

1 - SCOPE ....

2 - GENERAL APPROACH ....

2.1 - DEFENCE IN DEPTH PRINCIPALE - SAFETY FUNCTIONS ....

2.2 - DESIGN IMPROVEMENTS BY INTEGRATION OF EXPERIENCE FEEDBACK & USE OF PSA

2.3 - SAFETY DEMONSTRATION ....

3 - GENERAL TECHNICAL PRINCIPLES ....

3.1 - QUALITY OF DESIGN, MANUFACTURING, CONSTRUCTION AND OPERATION ....

3.2 - REDUCTION OF FREQUENCY OF INITIATING EVENTS ....

3.3 - IMPROVED PLANT TRANSCIENT BEHAVIOR ....

3.4 - REDUNDANCY AND DIVERSITY ....

3.5 - ACTIVE AND PASSIVE SYSTEMS ....

3.6 - INTEGRITY OF THE PRIMARY CIRCUIT ....

3.7 - MAN MACHINE INTERFACE ....

3.8 - QUALIFICATION OF COMPUTERIZED SYSTEMS ....

4 - USE OF PSA ....

5 - OPERATION

5.1 - OCCUPATIONAL EXPOSURES ....

5.2 - RADIOACTIVE RELEASES AND WASTE ....

6 - INCIDENTS AND ACCIDENTS ....

6.1 - ANALYSE TO BE PERFORMED ....

6.2 - RADIOLOGICAL CONSEQUENCES ....

7 - SEVERE ACCIDENTS AND CONTAINMENT BUILDING DESIGN ....

8 - EXTERNAL HAZARDS ....

<u>Appendix IV B : "Key issues of conceptual safety features" selected in 1994 for early decisions</u>

1 - SEVERE ACCIDENTS ....

2 - PRIMARY CIRCUIT INTEGRITY ....

3 - SYSTEM DESIGN & USE OF PSA ....

4 - RADIOLOGICAL CONSEQUENCES ....

5 - EXTERNAL HAZARDS

<u>Appendix IV C : Content of the "Technical Guidelines for the design and construction of the next generation of PWRs plants" adopted on October 26,2000 :</u>

INTRODUCTION AND SCOPE OF APPLICATION ....

A - PRINCIPLES OF THE SAFETY CONCEPT ....

A.1 - GENERAL SAFETY APPROACH ....

A.2 - GENERAL SAFETY PRINCIPLES...

B - CONCEPTUAL SAFETY FEATURES ....

B.1 - DESIGN OF BARRIERS ....

B.2 - SAFETY FUNCTIONS AND SYSTEMS..

C - ACCIDENT PREVENTION AND PLANT SAFETY CHARACTERISTICS.......

C.1 - REDUCTION OF THE FREQUENCIES OF INITIATING EVENTS ....

C.2 - REDUNDANCY AND DIVERSITY ....

C.3 - HUMAN FACTORS ....

C.4 - RADIATION PROTECTION OF WORKERS AND OF THE PUBLIC ....

D - CONTROL OF REFERENCE TRANSIENTS, INCIDENTS AND ACCIDENTS

D.1 - LIST OF REFERENCE TRANSIENTS, INCIDENTS AND ACCIDENTS ....

D.2 - SAFETY ANALYSIS RULES AND ACCEPTANCE CRITERIA ....

E - CONTROL OF MULTIPLE FAILURES CONDITIONS AND CORE MELT ACCIDENTS ....

E.1 - MULTIPLE FAILURES CONDITION ....

E.2 - PROTECTION MEASURES AGAINST CORE MELT ACCIDENTS................

F - PROTECTION AGAINST HAZARDS...

F.1 - PROTECTION AGAINST INTERNAL HAZARDS ....

F.2 - PROTECTION AGAINST EXTERNAL HAZARDS...

G - SYSTEM DESIGN REQUIREMENTS AND EFFECTIVENESS OF THE SAFETY FUNCTIONS ....

G.1- DESIGN OF THE FUEL POOL COOLING SYSTEM ....

G.2 - EFFECTIVENESS OF THE LEAKTIGHTNESS OF THE CONTAINMENT ....

G.3 - DESIGN OF INSTRUMENTATION AND CONTROL ....

G.4 - USE OF TECHNICAL CODES ....

Appendix V : Active or passive systems – the EPR approach

The approach described hereafter is the summary of options considered during the conceptual design phase of the EPR.

1 - DEFINITION OF "PASSIVE"

The IAEA definition of a passive component is: "a component which does not need any external input to operate. It may experience a change in pressure, temperature, radiation, fluid level and flow in performing its function. The function is achieved by means of static or dormant un-powered or self-acting means". The associated definition of a passive system is :"a system which is composed of passive components and structures" (see IAEA 623-I3-TC-633 "Description of passive safety-related terms).

The so-called "passive reactors" being developed are not strictly based on this definition, but more on a definition proposed by EPRI: "passive systems: systems which employ primarily passive means (i.e. natural circulation, gravity, stored energy) for essential safety functions - contrasted with active systems. Use of active components is limited to valves, controls and instrumentation" (see EPRI ALWR Requirements document).

It was passive features according to the EPRI definition that were included in the review at the beginning of the EPR conceptual design phase.

2 - CRITERIA TO ASSESS PASSIVE SYSTEMS

Passive features or systems should be subject to a systematic assessment with respect to the criteria of simplicity of design, impact on plant operation, safety and cost. The first two assessment criteria concern, in more general terms, simplicity.

Firstly, the design should be simplified, or at least not complicated by the implementation of passive features. In this context, proven technology of the components employed is required.

Furthermore, the degree of passivity should be investigated: where should a proposed solution rely on active equipment like valves or on active auxiliary systems like cooling or ventilation? Also, the overall system configuration should be simplified. If possible, an active system should be removed, or at least simplified by the implementation of a passive system. In addition, the overall system configuration should be simplified. An indicator of simplification could be a reduced requirement for system interconnections.

Secondly, the operation of the plant and of the passive system should be simple. Normal operational modes like power operation, start-up, shutdown, refuelling, and maintenance should not be affected by the passive system. Spurious actuation of passive systems would have to be investigated, as well as the possibility to detect it and to take straightforward recovery actions to avoid undue consequences on overall plant operation. The operation of the passive system itself should also be simple: this includes initiation which should be based on plant status and not on a perhaps difficult diagnosis of an accident scenario, as well as system operation (e.g. the need for adjustment of operational modes as a function of plant status or operating situation should be avoided).

As a rule, passive features chosen for implementation should be inspectable and have in-service testing capability with the testing mode being as close as possible to the operational mode of the system.

The last two assessment criteria concern safety and cost. As already mentioned, the implementation of a passive system should allow clear safety and economic advantages.

New accident scenarios should not be introduced by passive systems. The systems should fit in with the well-proven defence in-depth concepts and allow an extended time response to incidents or accidents. The incident consequences should not be aggravated by operation of the system. Furthermore, the multi-barrier concept (strong reactor coolant pressure boundary, control of containment leakages by double containment) used in French and German PWR designs should not be weakened by the introduction of passive systems.

3 - Passive features investigated for EPR

Some of the passive features that have been included in the design of EPR are described in Section 4.2.4 of this sub-chapter. In addition to these features, about twenty passive features were evaluated during the conceptual design phase, which are briefly described in the subsequent paragraphs.

3.1 - Passive high pressure residual heat removal system

The objective of this system is to remove the residual heat by primary side cooling in events where existing designs rely on secondary side cooling, to allow replacement of the emergency feedwater system (EFWS).

The primary water flows by natural circulation through the RHR heat exchanger located in an elevated water filled pool. The RHR heat exchanger is cooled by the pool water which evaporates into the containment. A containment cooling system becomes necessary or alternatively the pool must be cooled by an active cooling system. Active measures are required such as opening of valves for RHR system flow and start of heat removal from the pool or containment. The main results of the assessment of this system were the following:

- Flow rate through the RHR system depends (a) on the elevation between levels of the reactor coolant system (RCS) loops and the RHR heat exchanger and (b) on diameter of RHR pipes.
- The concept would lead to a significant extension of class 1 equipment.
- Installation of water pool including RHR heat exchanger at about the same level as the operating floor and assuming that more than one train, including pool, would be necessary, would lead to a complex arrangement of the reactor building.
- An operational system would also be necessary to bring the plant to cold shutdown conditions for refueling.

Although the passive RHR system presents the potential advantage of easy operation, it was not retained for the EPR because it failed to pass the criteria of design simplicity, safety improvement and cost reduction in the selection criteria.

3.2 – Safety condenser

The objective of this concept is to implement an autonomous, self-fed secondary-side residual heat removal system.

The main element of the system is the safety condenser itself, located outside the containment and connected to the steam generator on the steam side and on the water side, and the demineralised water pool, which is connected to the shell side of the safety condenser.

During normal plant operation the system would be on standby and separated from the SG by closed isolation valves in the condensate line. The valve in the steam supply line is locked in the open position, so that the condenser is full of cold condensate on the tube side and is at main steam pressure. On the shell side, the condenser is partially filled: the closed control/isolation valve prevents the inflow of demineralised water from the demineralised water pool. To start up the system on a safety demand, the redundant, diverse condensate drain valves and the isolation valve in the demineralised water supply system are opened and the load controller activated. Draining of the secondary side of the condenser exposes heat transfer surface; heat transfer from the steam generator to the condenser takes place when level on the tube side falls below that on the shell side. The cold condensate flowing from the condenser into the SG absorbs energy, before heat removal by the condenser begins. After the system run-up time, which is governed chiefly by the draining characteristic of the condenser, cooling begins. This is achieved by the admission, via the redundant, diverse control stations, of demineralised water from the demineralised water pool, which is at a higher static head. The result is evaporation to the atmosphere which acts as a heat sink.

The power for the valves required in normal operation and to ensure operation under emergency conditions is provided by a battery-backed supply. Since only a low electrical power is required, a grace period of several hours is available for restoring the function of any AC generators which may have failed.

Although the safety condenser concept presents potential reduction in activity release in the case of a SGTR, this concept was not retained for the EPR because it failed to pass all the selected criteria. Specifically, the system does not meet simplicity and cost criteria.

3.3 – Passive EFWS

The objective of this system is the same as that of the safety condenser. The heat exchanger and the demineralised water pool are combined in a single component instead of two separated components.

In order to avoid elevated storage of large inventory of water, an emergency feedwater tank under nitrogen or compressed air over-pressure, located at ground level, allows replenishment of the passive condenser as and when required, to make up for water evaporation. This concept was not retained for EPR as it failed to meet the simplification, operation and cost criteria.

3.4 – Secondary side residual heat removal and passive feed

The objective of this system is to remove residual heat from the core for events such as station blackout and complete loss of feedwater, by providing a passive mean to supply water to the steam generators (SG).

Elevated demineralised water pools, large enough to supply the SGs for several hours (station blackout duration), provide flow by gravity once the associated control valves have been opened and after closure of the SG main steam isolation valves. The cooldown is performed by steam release to the atmosphere through dedicated relief valves.

The main drawback of this concept is that the elevated pools must be protected against external events, particularly earthquakes. Movements of large inventories of water and the design of their supporting structures are major safety and cost challenges.

This concept was also not retained for EPR because it did not pass any of the four categories of evaluation criteria.

3.5 – Medium-head safety injection by accumulators

The objective of this system is to simplify the safety injection system (SIS) without reducing the safety level below that on existing plants. The idea was to delete the medium head safety injection (MHSI) pumps so as to reduce the SIS cost, to require reduced maintenance and to simplify operation of the system.

In order to fulfill the MHSI functions it is necessary to provide for an automatic depressurization system and high pressure accumulators. Potential difficulties arose during the assessment non-LOCA events with this concept. It was found that the management of steam generator tube rupture (SGTR) accidents would have to be reconsidered, and questionable operating modes were discovered.

The concept was dropped because it was found to lead to increased costs with respect to conventional active safety injection systems, thus failing to meet the primary objective.

3.6 – Gravity-driven low-head safety injection from tank/sump by primary system depressurization

The objective of this system is to provide an efficient ultimate back-up for injection of water at low pressure for long term reactor cooling. The RCS is flooded with water above the loop level and water flows by gravity from sumps, through check valves, into the reactor vessel. Decay heat is removed to the containment atmosphere by evaporation of the flood water. Steam produced inside the containment condenses on cooled surfaces of a containment cooling system and the condensate flows back to the RCS.

Active measures are required e.g. : opening of isolation valves, opening of RCS discharge and feed line from the sumps, start-up of the heat removal system in the containment.

The principle results of the assessment of this system were the following:

A large amount of water is necessary to flood the RCS, the amount depending on the reactor building lay-out. For the French 4 loops plants with cylindrical prestressed concrete containment, this volume may vary between 4700 m3 to more than 10000 m3.

Large diameter of discharge line(s) and small flow resistance check valves are necessary to allow gravity flow to the reactor vessel. Spurious opening of valves in the discharge line(s) would have to be avoided. Additional connections to the RCS are required for discharge line(s) and feed line (s). A full scale test to verify the concept would be difficult and extremely costly.

Although the passive low head safety injection system presents advantages, such as providing a back-up to low head safety injection pump and avoiding long term recirculation outside the containment, it was not retained for the EPR because it failed to meet the criteria of design simplicity, safety improvement and cost reduction.

3.7 – External Cooling of Metal containment

For metal containment structures, a concept in which heat removal is ensured by conduction through the containment wall is feasible in principle. Inside containment heat transfer would be by natural convection in the containment atmosphere and condensation on the containment wall inner surface. For outside containment heat removal, several alternate cooling schemes can be envisaged. However a completely passive concept, using natural circulation air cooling, would only be possible for small unit sizes and in the long term, after decay heat is sufficiently reduced. Thus, additional means based on water spray on the outside containment surface is required at least in the short term. For the larger unit size of the EPR, such water-circulation assisted outside cooling would be required even in the long term. Use of water cooling on the containment outside surface, without evaporation and based on an active cooling circuit with a pump and heat exchanger, would allow retention of a double containment barrier, which would not be possible in case of a natural air circulation cooling mode. However, in such a containment heat removal concept, the passive means of heat removal is provided only on the inside of the containment. Furthermore, the heat transfer capacity by condensation on the inner containment surface in the presence of non-condensable gases is limited and, for the larger EPR unit size, could result in elevated containment pressures (several bar), in the medium term following an accident.

For these reasons, as well as because the EPR uses a concrete rather than a steel containment concept, this option has not been retained for EPR.

3.8 – Sump cooler with passive cooling chain

The objective of this concept is to remove decay heat following a LOCA by natural circulation from the reactor building sump via submerged coolers and a secondary cooling system to the atmosphere.

Like many other passive concepts, opening of valves is necessary to start operation of this system.

Additional measures to transfer the heat from the containment sump were estimated to be necessary during the evaluation of this concept. A large heat transfer surface for sump cooler (a minimum of 1000 m²) was found to be required.

The passive sump cooling feature was dropped because the large height difference required between the ultimate heat sink and the sump cooler to secure natural circulation (a minimum of 20 m) would lead to unacceptable costs.

3.9 – Containment condenser coolers

The objective of this system is to provide a passive means, at least inside the reactor building, for removing decay heat in the long term after a severe accident, in order to avoid the internal pressure exceeding the containment design pressure.

Steam, driven by natural circulation, condenses on the outside surface of coolers. Cooling water circulates inside the coolers. The cooling system is active and located outside the containment.

The major drawbacks of this system, in comparison to existing spray systems, are the following:

- The heat transfer and containment pressure reduction capability are strongly dependent on the presence of non-condensable gas and on general convection movements inside the containment,

- The condensers must be located in the upper part of the containment where hydrogen is likely to accumulate. They constitute potential hydrogen traps, thus reducing their heat transfer capability and increasing the risk of explosion,
- A large amount of room would be required above the operating floor which is a congested area during maintenance and refueling,
- The condensers are of no help in reducing source term outside the containment because they have no effect on aerosols,
- The system decreases containment pressure slower than a spray system.

However, this system offers several advantages, the major one with respect to a containment spray system, being that it avoids recirculation of highly radioactive water outside the containment after a severe accident. The containment condenser coolers might therefore be reconsidered as a solution to one of the key severe accident challenges: how to remove heat from the containment building without circulating any fluid through its walls and without impairing its leak tightness?

Many engineers have come around to the idea that hybrid systems combining active and passive features represent an attractive alternative to existing designs. This view is supported by the utilities which contributed to the development of the EPR design. They consider that simplicity, reliability and less complicated control and automation are of greater safety importance than use of purely passive features.

4 - CONCLUSION

The environment of the nuclear power industry is changing throughout the world, both due to economic factors and greater impact of public perception on nuclear plant designs. It has become more important than ever to address both the technical and public perception issues in the design of nuclear reactors.

The design process begins by defining the intended plant size for future plants. Current experience in France, Germany, and the rest of the world indicates that the new plant sizes, at least for the foreseeable future, will be large, in the 900 MWe and above range. The preference for larger plants sizes is based on proven operating experience, and the economic advantages of size, based on available proven technology.

The EPR has been developed to address all issues of safety, public perception and economics. The experience to date in both France and Germany on standardization of plant design has been a major factor in the overall EPR design.

Although the current standard designs have performed well, evolution of the design, using better active and passive features lead to improvements in overall plant safety and economics. Additionally, the development of a standard plant that is nearly fully designed prior to obtaining a construction license, allows for the concentration of large engineering efforts in R&D, design, manufacturing practices, maintenance tooling and procedures to meet market demands for safety, availability and economy.

In general, the majority of passive features proposed thus far are still unproven through test or operation. As such, the features remain economically unjustified or liable to lead to plant complications that may actually degrade rather than enhance safety. For these reasons, EPR designers have not embraced a large number of new passive features for use in the EPR. They have favoured the approach of increasing the reliability of active safety features, for example by supplementing the emergency power supply needed for powering the active safety feature with diversified diesels in order to ensure that the safety functions will be achieved with a very high level of reliability.