CHAPTER H: INTERNATIONAL PRACTICE

1. INTRODUCTION

This Chapter reviews the EPR design assessments that have been carried out by international regulatory authorities and compares EPR design principles with current international standards. It has been prepared to address clauses 2.14 and 2.15 of Step 2 of the design acceptance process in the HSE guidelines for pre-licensing of new reactor designs in the UK.

The chapter considers the reviews carried out by the French Regulator for the Flamanville 3 EPR, the Finnish Regulator for the Okiluoto 3 EPR and the USNRC assessment for a generic EPR design. A comparison is also made between EPR design features and design features of the Sizewell B PWR in the UK. Some inputs on how international experience feedback has been taken into account in EPR design are presented. Finally a review is carried out which confirms that the EPR design meets the internationally agreed WENRA reference levels and the IAEA and EUR design safety standards.

2. REVIEW OF DESIGN BY FRENCH REGULATOR

Following the decision in 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 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).

The general design objectives laid down by the Safety Authorities were:

- to apply an evolutionary design process making maximum use of design and operating experience from existing reactors;
- to obtain significant improvements of safety in all the levels of defence in-depth by:
    - minimizing the dose to personnel and radioactive waste production in normal operation;
    - reducing the probability of accidents;
    - reducing the radiological consequences to the environment in the event of an accident.

A process of design development and optimisation then followed, conducted under the oversight of the French and German Safety Authorities and their technical support organisations (see Chapter C of this Head Document). This process resulted in the development of the design of the Flamanville 3 (FA3) EPR unit. The French Regulator completed a technical examination of the FA3 Preliminary Safety Report in September 2006. On conclusion, the French Regulator publicly stated their opinion, as follows:

- that no point was identified that called into question the achievement of the safety objectives defined in 1993,

- that satisfactory account had been taken of safety experience gained from reactors currently in operation,
- that the design improvements relating to industrial and public safety, compared to design of currently operating reactors, were acceptable,
- that no questions had been raised in respect of the design of the primary and secondary circuits,
- that no significant non-radiological industrial risk to the public or local environment had been identified,

Noting that the Flamanville 3 design had been subjected to a much broader and thorough examination than previously realized on French reactors at the stage of the Preliminary Safety Report, and that experts from several European countries had contributed to the examination, the French Regulator delivered a positive opinion on the project.

Sections 3 and 4 of Sub-chapter C.1 of Volume 1 of this pre-licensing submission provide details of the EPR design development process, and the specific design improvements that have been adopted.

EPR safety aspects were assessed by the French Safety Authority during the different phases of the EPR design described in Chapter C.3 of Volume 1 of this Fundamental Safety Overview. The “Technical Guidelines” produced at the end of the Post Basic Design Optimisation Phase can be considered as the synthesis of the different topics assessed.

All the recommendations of the French Regulator, from the beginning of the EPR design until the issuing of the Flamanville 3 construction license (DAC), were underwritten by studies performed by its Technical Support Organisation, IRSN (Institute for Nuclear Safety and Radiation protection). Over a fifteen year design period, about ninety EPR design assessment reports were issued by IRSN comprising more than six thousand pages of detailed technical analysis. These reports were tabled at meetings chaired by the French Standing Group for Nuclear Reactors (GPR), an independent advisory body established to support the French Regulator, consisting of scientists and engineers from France and other European countries and the USA (see Chapter C.2 of Volume 1). Conclusions of the GPR with regard to the IRSN recommendations were then transmitted to French Regulator by letter. Table 1 lists the design assessment reports produced by the IRSN over the review period which were tabled at meetings with the GPR.

Table 1: IRSN Design Review Reports

Report Topic	**Date**
IPSN/GRS analysis of NPI General safety design bases	Apr-92
Comments on NPI report on PSA	Feb-93
IPSN/GRS proposal for a Common safety approach for future PWR	Mar-93
IPSN/GRS analysis of NPI Radiological LOCA analysis methodology	Mar-93
External hazards	Mar-94
Severe accident approach and associated radiological consequences	Apr-94
System design and use of PSA	Jun-94
Integrity of the reactor coolant boundary	Jun-94
Radiological consequences of design basis accidents	Oct-94
Radiological consequences of severe accidents	Nov-94
Containment design (preliminary comments)	Jun-95
Format and contents of ETC s	Jun-95

Secondary side overpressure protection	Sep-95
Radiation protection during normal operation	Sep-95
System design issues : Redundancy - SFC - Secondary side heat removal - Electrical power supply - Containment bypass - SGTR - IRWST	Sep-95
Protection against internal hazards : overall approach	Sep-95
Break preclusion implementation on main coolant lines	Apr-96
Safety injection mode and 2A-break LOCA analysis rules	Apr-96
R&D program – preliminary review	Apr-96
Protection against internal hazards	Aug-96
R&D program (updated review)	Dec-96
Secondary side overpressure protection (with two safety valves)	Dec-96
Feedback from experience	Mar-97
General safety requirements related to system design - overall approach - classification requirements - rules for accident studies and for systems design	Mar-97
Preliminary PSA	Mar-97
Implementation of safety requirements to ten systems - SIS - IRWST - RHRS - EFWS - CCWS- ESWS - FPCS - CVCS - AVS - Electrical power supply	Mar-97
Status of GPR/RSK recommendations in January 1997	Mar-97
Analysis of emergency core cooling mode	Feb-98
Severe accident overall approach and related design features	Sep-97
General design of the primary containment	Sep-97
Structure of the guidelines	Sep-97
Comparison of methods used to calculate the radiological consequences of accidents	Sep-97
Status of GPR/RSK recommendations in December 1997	Dec-97
Protection against external hazards	Feb-98
Preliminary core design	Feb-98
Structuring GPR/RSK recommendations as guidelines	Mar-98
general safety requirements related to system design (updated review) - overall approach - classification concept - scope of events and accident studies - rules for system design	May-98
Shutdown states	May-98
First draft of parts A and B of the technical guidelines	Aug-98
General design of the primary containment (updated review)	Oct-98
Man machine interface	Oct-98
Instrumentation and control	Oct-98
Secondary side overpressure protection (updated review)	Oct-98
Status of GPR/RSK recommendations in December 1998	Dec-98
Proposal of guidelines for the calculation of radiological consequences of severe accidents	Mar-99
Waste reduction and dismantling	Mar-99
System design and accident studies issues - heterogeneous dilution - SFC and stuck rod - barrier classification - safety approach of the NAB	Mar-99
ETC issues of electrical equipment and handling devices	Mar-99
Status of the technical guidelines in march 1999	Mar-99
Severe accident issues	Jun-99

Confinement function	Jun-99
ETC issues of I&C equipment and fire protection	Jun-99
System design and accident study issues (continuation) - Passive SFC - FPCS - Boration systems - CCWS - EFWS	Jun-99
Status of the technical guidelines in June 1999	Jun-99
Proposal of guidelines for the calculation of radiological consequences of severe accidents (revision A)	Jul-99
GPR - German experts recommendations status in August 1999	Aug-99
Status of the technical guidelines in September 1999	Sep-99
Containment design and liner implementation	Oct-99
Radiation protection : ALARA approach	Oct-99
System design issues (continuation) : Safety classification - Equipment qualification - Effluent treatment systems	Nov-99
ETC issues of civil works and ventilation systems	Oct-99
Shutdown states (updated review)	Nov-99
Accident studies and level of power	Feb-00
PSA	Feb-00
Confinement function (updated review)	Feb-00
Heterogeneous boron dilution	Feb-00
System design issues (continuation) : FPCS - CHRS	Feb-00
Secondary side break preclusion	Feb-00
Severe accident issues (continuation)	Feb-00
BDR issues and EPR commitments - earthquake - Links between external and internal hazards - radiation protection - Man machine interface - waste, effluents and dismantling	Mar-00
GPR - German experts recommendations status in mars 2000	May-00
Status of the Technical Guidelines in July 2000	Jul-00
Draft of the technical guidelines in July 2000	Jul-00
Remarks on the draft of technical guidelines	Oct-00
Technical Guidelines for future PWRs	Nov-00
Barrier classification - confinement of peripheral buildings - systems	Jul-02
Human factors	Jun-03
PSA	Jun-03
Design of the fuel pool - list of PCC and RRC-A	Jun-03
I&C - Containment with steel liner - Radiation protection - Finnish safety approach and review of the YVL safety guides	Jun-04
Principles of computerized operation - Design of the core catcher - Heterogeneous boron dilution - Design of the safety injection system - Multiple failures of non seismic equipment - Combination of hazards	Oct-04
Radiation protection - Equipment qualification - Break preclusion on MSL - Containment bypass situations - Pumping station and diversity of UHS - Safety requirements for civil work design - Extreme hot temperature situations	Jun-05
RHR break - I&C - Emergency station - Equipment hatch - core catcher - preventive maintenance in power	Nov-05
Equipment qualification - Breaks > 50 mm in shutdown states in RB - Prevention of fuel melting in fuel pool - Radioactive releases and wastes - External flooding	Jan-06

Protection against hazards - Probabilistic safety studies - Water intake clogging risk in IRWST - Principles of computerized operation - Waste zoning - Site related topics - Other topics	Jun-06

Areas that subject to the most in-depth regulatory assessment were linked to the evolution of the PWR design and EPR design features that were novel compared with currently existing plants, such as:

- design against severe accidents: demonstration of the “practical elimination” of sequences leading to large radioactive releases and mitigation of core melt sequences and the behaviour of the core catcher,
- containment design: demonstration of the ability of the internal structures to withstand loads resulting from accident situations (in particular due to hydrogen detonation in case of severe accidents) and ability of the external structure to withstand loads due to a wider range of external hazards (in particular due to the crash of a large commercial airplane),
- exhaustiveness of the safety case: increased number of events (single initiating events/ accidents and multiple failures sequences) and hazards (internal and external) considered in the plant design, and analysis of events and accidents in all reactor states (from the cold shutdown states up to the full power state).
- I&C: assessment of the new technology associated with a four safety train design and principles of computerized operation.

3. REVIEW OF DESIGN BY USNRC

3.1. INTRODUCTION

The U.S. EPR is currently in the advanced stage of a preliminary review by the U.S. Nuclear Regulatory Commission (NRC), leading up to a formal application for design certification that will be submitted late in 2007. This brief paper provides an overview of the NRC’s design certification process, describes the progress to date in the pre-application review, and discusses plans for the completion of pre-application activities and transition to the design certification review. In addition, technical issues that have been or are expected to be significant in terms of their impact on the design certification review are highlighted.

3.2. NEW US LICENSING PROCESS

The current US operating fleet of commercial nuclear power plants was licensed using a "two-step" construction permit/operating license process. The need for two separate licensing proceedings sometimes led to long delays between completion of construction and plant operation, and in a few cases, a plant that was essentially complete was never put into operation. Consequently, in 1989, the NRC established a new "one-step" licensing process, whereby a combined construction permit and operating license (COL) could be issued. The COL allows plant operation following the completion of construction, provided that the plant owner and the NRC confirm that the plant, as constructed, conforms to the design as licensed, by means of an agreed-upon set of inspections, tests, analyses, and acceptance criteria (ITAAC).

In addition to the COL process, the NRC established two other new processes: Early Site Permit (ESP) and design certification (DC). An ESP allows a prospective plant licensee to get approval for a site in advance of applying for a COL. The DC process involves NRC review and approval of a standardized, "generic" plant design. Following the technical review of the plant, the essential attributes of the design are "certified" by incorporating them into a rule that becomes part of the NRC's compendium of regulations. An application for a COL can then reference the DC rule as part of the application. All technical issues associated with certification of the generic design (i.e., non-site-specific) are considered as "resolved," and are not subject to reconsideration during the COL review.

To facilitate the review process, the NRC encourages prospective applicants to engage in "pre-application" discussions with the agency, and to submit documentation to familiarize the NRC staff with the plant design, particularly with regard to safety features, and to identify key technical issues that may require substantial NRC effort to resolve. The NRC also accepts topical reports, providing detailed technical information, for review during the pre-application process, to facilitate its assessment of the DC documentation. AREVA is currently involved in the pre-application process leading up to the NRC's DC review of the U.S. EPR. The following discussion summarizes the history of the pre-application process to date, and provides a look ahead at plans for the remainder of 2007.

3.3. U.S EPR DESIGN CERTIFICATION PRE-APPLICATION REVIEW

AREVA formally initiated the pre-application process for the U.S. EPR design by a letter to the NRC, dated February 8, 2005. In that letter, AREVA outlined a two-phase pre-application process extending over nearly three years. The first phase, extending over the remainder of 2005, would involve a series of meetings between AREVA and the NRC staff, approximately one per month between March and December, to discuss aspects of the U.S. EPR design and associated analytical methods for safety analyses. These meetings included one trip by members of the NRC staff to visit AREVA facilities in Europe. AREVA also committed to submit a Design Description Report (DDR) for the U.S. EPR in August 2005, for the NRC's information. The second phase of the pre-application review was projected to begin in early 2006, ending with the submission of AREVA's DC application in December 2007. At that time, AREVA expected to submit only four topical reports for NRC pre-application review. Additional meetings were also proposed in Phase 2, but the topics were left to be determined.

Rather than the 10 meetings proposed in the letter, AREVA and the NRC met only four times during 2005 and once during January 2006, including the proposed NRC trip to Europe. The DDR was submitted to the NRC in August, as scheduled. Table 2 shows the meeting topics and dates for Phase 1.

Table 2. U.S. EPR Phase 1 Pre-Application Meetings

Meeting Topic	Date
Design Overview and Pre-Application Plan	3/24/2005
Plant Design Bases and PRA	7/21/2005
NRC Visit to European Facilities	10/21/2005
Analytical Methodology Overview and Severe Accidents	11/2/2005
Plan for Phase 2 of Pre-Application Review	1/10/2006

In the meeting of January 10, 2006, AREVA and the NRC agreed that Phase 1 of the pre-application process had been completed, and that Phase 2 could begin. Subsequently, AREVA sent a letter, dated February 3, 2006, to the NRC, proposing an extensive schedule of about 15 meetings during calendar year 2006. The original proposal for four topical reports, as described in the February 2005 letter, was expanded substantially. About 10 topical and technical reports were proposed for submission in 2006, with an additional 6 reports identified for submission in 2007. As suggested by the NRC, several of the meetings were scheduled to precede the submission of topical reports, to review the proposed content of the report and get NRC feedback to permit the staff's concerns to be addressed in the report.

Overall, the meeting and report schedules were met, for the most part, as originally proposed with 15 meetings (several covering multiple topics) and 11 topical or technical reports. Tables 3 and 4 summarize the 2006 meetings with (not including the January 10, 2006, meeting noted above) and reports submitted to the NRC.

Table 3. U.S. EPR Phase 2 Pre-Application Meetings in 2006

Meeting Topic	Date
Use of Design Acceptance Criteria in U.S. EPR DC Application	2/23/2006
Overview of U.S. EPR Compliance Evaluation	3/29/2006
I&C (RPS Philosophy and Design Concepts)	4/27/2006
Technical Specifications	5/31/2006
• Fire Protection and Train Separation Criteria • Electrical System Design	6/20/2006
Security	7/19/2006
Severe Accident Testing Program and Severe Accident Topical Report Pre-Submittal	7/25/2006
Code Applicability Topical Report Pre-Submittal	8/1/2006
I&C (Safety System and Operational I&C Architecture)	8/31/2006
• Unique Design Features • Containment Analysis • In-core and Nuclear Instrumentation Design Pre-Submittal	10/25/2006

Meeting Topic	Date
PRA Analysis Tools	10/24/2006
AV42 Priority Control Module Pre-Submittal	8/31/2006
Equipment Qualification Program Pre-Submittal	11/29/2006
CHF Correlation Topical Report Pre-Submittal	11/14/2006
Human Factors Program Report Pre-Submittal	12/7/2006

Table 4. U.S. EPR Phase 2 Pre-Application Topical and Technical Reports Submitted to the NRC in 2006

Report Topic	Date Submitted
Codes and Methods Applicability Topical Report	8/11/2006
Quality Assurance Plan Topical Report and DCD Subsections 17.1, 17.2 and 17.3	9/22/2006
Piping Analysis and Pipe Support Design Topical Report	9/29/2006
Severe Accident Evaluation Topical Report	11/1/2006
Detailed Design Description Report on Unique Design Features	11/30/2006
CHF Correlation Topical Report	11/30/2006
AV42 Priority Control Module Topical Report	11/28/2006
Software Program Topical Report	12/21/2006
Environmental Qualification Program Report	12/21/2006
PRA Methods Report	12/15/2006
In-core and Nuclear Instrumentation Design Report	12/15/2006

AREVA is beginning to receive the NRC's questions, formally called Requests for Additional Information (RAIs), on some of the earliest reports, and is preparing responses and revisions to the reports (if necessary), as appropriate. Follow-up meetings with the NRC to discuss RAI responses will also be scheduled.

Meetings and submission of additional reports will continue through 2007 as Phase 2 of the pre-application process draws to a close and the DC application materials are completed for submission to the NRC by December 14, 2007. Tables 5 and 6 show the pre-application activities planned for this calendar year.

Table 5. U.S. EPR Phase 2 Pre-Application Meetings in 2007

Meeting	Date (Actual or Planned)
Instrument Setpoint Methodology Topical Report Pre-Submittal	1/30/2007
Digital Protection System Design Topical Report Pre-Submittal	3/1/2007
DC Planning	4/2007 or 5/2007
I&C Diversity and Defence-in-Depth Topical Report Pre-Submittal	5/2007
PRA Methods Report Post-Submittal	5/2007
Equipment Qualification Program Report Pre-Submittal	5/2007
Technical Specifications	6/2007

Meeting	Date (Actual or Planned)
Fuel Assembly Mechanical Analysis Topical Report Pre-Submittal	8/2007
Setpoints Safety Analysis Methods Topical Report Pre-Submittal	10/2007
Reactivity Insertion Accident Topical Report Pre-Submittal	10/2007
DC Application Pre-Submittal	11/2007

Table 6. U.S. EPR Phase 2 Pre-Application Topical and Technical Reports Planned or Submitted to the NRC in 2007

Report Topic	Date Submitted (Actual or Planned)
Human Factors Program Topical Report	1/31/2007
Instrument Setpoint Methodology Topical Report	3/30/2007
Realistic LBLOCA Topical Report	3/30/2007
Digital Protection System Design Topical Report	3/30/2007
Accident Analyses Results (Quick Look) Summary Report	6/1/2007
I&C Diversity and Defence-in-Depth Topical Report	6/29/2007
Fuel Assembly Mechanical Analysis Topical Report	9/28/2007
Setpoints Safety Analysis Methods Topical Report	12/14/2007
Reactivity Insertion Accident Topical Report (RCCA Ejection Methods)	12/14/2007

3.4. KEY TECHNICAL ISSUES

The subjects of the topical and technical reports shown above are representative of the key technical issues on which the NRC will focus during the design certification review. Because of the U.S. EPR's "evolutionary" approach to safety, relying principally on active safety systems, the operational characteristics of the engineered safety features are, in general, familiar to the NRC. The one unique safety feature of the U.S. EPR is the elimination of the high-head safety injection (SI) system, which has been replaced by a medium-head SI system and a partial depressurization capability. The ability of AREVA's safety analysis codes to model the operation of this system will likely be a key element of the NRC review. The NRC will also want to ensure that AREVA has adequate data to demonstrate the applicability of thermal-hydraulic models (e.g., critical heat flux) to the U.S. EPR's 14-ft fuel length.

While severe accidents are not included within the design basis for U.S. plants, and structures, systems, and components (SSCs) for severe accident mitigation are not required to meet the same standards as safety-related (design basis) SSCs, severe accident performance and probabilistic risk assessment (PRA) modeling for new reactors is receiving increased attention from the NRC. Another significant issue that will affect all advanced reactor designs is the use of digital instrumentation and control (I&C) systems, along with the associated human factors engineering (i.e., control room design). There are no operating plants with fully-digital safety I&C systems, and the NRC's approach for reviewing these systems is still evolving. The review of I&C systems and control room design may be a critical path item in the design certification review.

3.5. THE MULTINATIONAL DESIGN EVALUATION PROGRAM

The NRC has established a broad collaborative program with regulatory agencies in other countries. This Multinational Design Evaluation Program (MDEP) provides for the exchange of technical assessments of issues of common interest between regulators. Each regulator can then use the technical evaluation in making its own regulatory decisions on the issues, consistent with its nation's policies. The EPR has been designated as the lead plant for implementation of Stage 1 of the MDEP, in which the NRC is working with the Finnish (STUK) and French (ASN) regulatory agencies. (Stage 2 of the MDEP, involving broader multilateral cooperation on a range of generic elements, is also underway; the third and final stage is for the future.) The NRC has expressed a willingness to consider expanding Stage 1 to other plant designs and/or additional regulators. Although the NRC and the UK regulatory authorities have discussed cooperation in advanced plant reviews, it is not known at this time if the British regulators (or others) will ultimately join the EPR MDEP effort.

3.6. NRC DESIGN CERTIFICATION REVIEW

After AREVA submits its DC application, the NRC will conduct an acceptance review of approximately 30 days' duration to determine whether the information in the application is sufficient for the agency to initiate its technical review. If the NRC determines that the application is acceptable, it will issue a letter to AREVA establishing its proposed review schedule, which will comprise the technical assessment of the plant design, culminating in the issuance of a Final Safety Evaluation Report (FSER), followed by the formal rulemaking process. Issuance of the design certification rule and its incorporation as an Appendix to the NRC 's rules in Title 10, Part 52 of the *Code of Federal Regulations* (10 CFR Part 52) completes the design certification.

Although the NRC's official schedule will not be available until after the acceptance review is completed, AREVA's objective is to facilitate the process to permit the NRC to complete the technical review and issue the FSER in approximately 24 months, followed by a 10-month rulemaking.

3.7. CONCLUSION OF USNRC DESIGN REVIEW

This sub-chapter has summarized the USNRC design certification process and the current status of AREVA's efforts in the pre-application process leading to the formal submission of an application for design certification of the U.S. EPR. To date, AREVA has conducted an extensive series of meetings with the NRC staff, focusing on significant technical elements of the plant design and associated analytical models, and has submitted a dozen topical and technical reports on these subjects. Additional meetings and reports are planned for much of the rest of 2007, prior to submission of the design certification application in December 2007.

4. REVIEW OF DESIGN BY FINNISH REGULATOR

4.1. LICENSING PROCESS FOR EPR IN FINLAND

A Construction Licence for the Okiluoto 3 (OL3) EPR was granted by the Finnish Government in February 2005. The main components of the Finnish licensing approach were:

- A political approval process in advance of the industrial decision process,
- An Environmental Impact Assessment performed before political approval which was decoupled from details of the different candidate reactor designs considered, but which was able to provide sufficient information to support of the political approval process,
- A well defined regulatory context,
- A feasibility study of all "candidate designs" to ensure no safety issues preventing compliance with the Finnish nuclear safety regulations existed.

The Finnish Government made a “Decision in Principle” in January 2002 which concluded that the construction of a new nuclear power plant in Finland was "in line with the overall good of society". This decision was ratified by the Finnish Parliament in May 2002.

The Environmental Impact Assessment (EIA) for the candidate designs was started in May 1998, and was completed in January 2000. It was considered by STUK, the Finnish regulatory body, that the EIA did not require detailed information on the specific plant type. The EIA used data from existing nuclear units adjusted to the safety requirements for a new plant. The EIA was done for two potential sites, both of which already contained operating nuclear plants.

To support the pre-licensing process, TVO, the applicant, reviewed with the potential reactor vendors the compliance of their designs with Finnish regulations and also considered construction issues.

The safety assessments carried out by TVO and the reactor vendors were presented to STUK. STUK concluded that all alternative designs mentioned in the application could probably be made to fulfil Finnish safety requirements, but none of the plants seemed acceptable as presented and some modifications would be needed in all designs.

After the statement by STUK had been issued, the 9-11 events took place, and the Ministry responsible for nuclear licensing asked STUK whether it would be possible to provide protection of the reactors against severe plane crashes. STUK in response issued new safety requirements with respect to external impacts, and concluded that it was feasible to meet them.

4.2. OUTCOME OF REVIEW BY FINNISH REGULATOR (STUK)

The Finnish process for nuclear safety regulation is described in numerous papers, and information is available on the STUK website. The regulatory process is based on well-established national and international practices, and Finnish safety requirements incorporate state-of-the-art developments in nuclear safety technology. 73 detailed regulatory guides (YVL), produced by STUK, are currently in force.

Assessment of the EPR concept against the YVL guides resulted in a small number of modifications being introduced specifically for Okiluoto 3.

The most significant modifications that were considered by STUK to be required by Finnish Licensing rules, were as follows:

- In spite of the application of the Break Preclusion principles, STUK required that account was taken of the mechanical consequences of a postulated guillotine break of the main RCS coolant pipework. Consequently anti-whipping devices will be installed in Olkiluoto 3.

- In spite of the measures implemented in the design to use diverse I&C platforms, it was required that failure of the digital I&C continued to be postulated in the design of Olkiluoto 3. The consequence was the implementation of a hardwired backup system to ensure plant shutdown in case total failure of digital I&C systems were to occur during a PCC-2 or a frequent PCC-3 event.

- In spite of the dedicated measures implemented to ensure heat removal from the containment after a low pressure core melt event, which are designed to ensure that the containment pressure remains below the design pressure, installation of a containment venting system was required in Olkiluoto 3.

- Application of the Finnish rules with regard to fire prevention and mitigation impacted on access rules, resulting in changes to some design features of the Olkiluoto 3 ventilation systems.

The modifications requested by STUK were reviewed by the GPR (see Section 2 above) on behalf of the French Safety Authorities, but none were recommended for implementation. As the proposed UK PWR is based on the Flamanville 3 EPR design approved by the French Safety Authorities, none of the Olkiluoto 3 modifications are therefore included in then UK design.

4.3. REGULATORY CONTROL DURING CONSTRUCTION

STUK is implementing regulatory controls which are based on a very detailed design review process and surveillance during manufacturing. It implies a one by one approval of the design documentation (Construction Plan, System description) and a large number of “hold points” in the manufacturing process.

In the context of a first-of-a-kind design where the detailed design had to be finalized in parallel with the construction, in combination with a loss of experience of some key sub-suppliers, some delays have occurred to the construction time schedule.

The next licensing milestone is issue of the operating license, scheduled for 2009. The operating license will be issued after a detailed review of documents including among others:

- A Final Safety Analysis report, containing accident analysis and topical reports based on actual systems, structures and components and a description of unit commissioning and operation.

- A Probabilistic Safety Assessment, containing PSA level 1 and 2 analyses.

- A quality assurance programme for operation

- Technical Specifications
- A summary programme for in-service inspections
- Physical protection and emergency response arrangements
- Arrangement of the necessary controls to prevent the proliferation of nuclear weapons
- Administrative rules
- Arrangements for environmental radiation monitoring

More generally the overall readiness of the plant to enter commercial operation will be verified and checked with a satisfactory completion of the commissioning and a proper training of the staff.

Commercial operation is scheduled to be achieved at the end 2010 – early 2011.

5. SUMMARY OF INTERNATIONAL ADAPTATIONS OF EPR DESIGN

EPR projects in different countries are based on the Nuclear Island design developed during the EPR Basic Design phase. However the design of the Turbine Island and Balance of Plant may differ more significantly from project to project.

The EPR Basic Design took into account French and German safety requirements and French and German utilities requests, in addition to the European Utility Requirements. Other country specific regulations and practices were not considered at that time, nor were specific expectations of utility customers outside France and Germany.

It is thus likely that some adaptations of the design will be needed for each project in order to meet country-specific regulations and practices, local site conditions and specific utility requirements. For example, although the general design of the EPR fire protection system and equipment is based on the requirements of the EPR Technical Code for Fire Protection, the need to comply with relevant local regulations, codes and guidelines may lead to differences in the implementation and design of fire protection measures

For each project, the objective is to accommodate any specific requirements and for subsequent projects to benefit from the experience feedback from preceding EPR projects, while maintaining, as far as possible, a common reference design across the various projects.

Table 7 provides a general overview of principal design differences between the EPR designs that are presently being built in Finland (Olkiluoto 3) and in France (Flamanville 3) and the EPR design to be submitted for Design Certification in the USA. The table is not intended to be exhaustive. The potential design variations will be considered in relation to the UK EPR design against UK principles of ALARP later in the pre-licensing process to confirm that the UK design complies with the ALARP requirement.

It is emphasised that the various concurrent projects regularly exchange information to control differences deriving from different national requirements, and to ensure that conflicting solutions are not implemented in different projects.

6. INTERNATIONAL EXPERIENCE FEEDBACK

In designing the EPR it was decided to follow an evolutionary approach: the advantage of basing an advanced design on operational experience from approximately 100 nuclear power plants in the world (Belgium, Brazil, China, France, Germany, Korea, South Africa, Spain) constructed by Framatome and Siemens was deemed by the designers to be very important.

In addition, experience feedback from other nuclear power plants has been reviewed and design features addressing the generic safety issues identified have been taken into account. The following examples illustrate this approach:

- SG tube integrity has been improved by the choice of Inconel 690 as the tube material to avoid corrosion cracking and intergranular attack. Denting and fretting are prevented by an optimized design of the internals and supports. In addition, the design provides accessibility to the tube bundle for inspection and maintenance.

- Overfilling of SG in case of SG tube rupture is avoided by the reducing the head developed by the MHSI pumps and by automatic initiation of fast RCS cooldown.

- ECCS sumps blockage: in order avoid blockage by insulation material and other debris, a staggered strategy for debris retention is adopted: this includes use of weirs and a trash rack for protection of openings in the heavy floor, large retention baskets with overflow space below the heavy floor opening and large screens with small meshes. The large screens with small mesh size above the sump pit have a robust construction to cope with increased head losses due to debris blockage and inclined subdivided sump screens are employed in order to facilitate filter cake detaching.

- In order to improve SG feedwater system availability, the EPR is equipped with a four independent train Emergency Feedwater System, each train being powered by a segregated diesel generator. In addition the plant design includes a Startup and Shutdown Feed System,

- Improved reliability for the power supply system: The Emergency Power Supply System is equipped with four separate and independent diesel generator units that are safety grade and they are automatically started by low voltage or low frequency signals. In addition the Station black-out power supply has two separate and independent diesel generators which are of a diverse design from the emergency diesel generators sets; they are also safety grade.

- Following a core melt accident, the containment integrity is ensured through design measures dealing with hydrogen detonation, direct containment heating, vessel lift, ex-vessel steam explosion, basemat (foundation raft) melt-through, containment pressurisation and containment leakage.

A systematic review has been carried out to confirm that the EPR design addresses generic issues identified in IAEA Tecdoc IAEA-TECDOC-1044, "Generic Safety Issues for nuclear power plants with light water reactors and measures taken for their resolution". A similar review in regard of NRC generic safety issues (NUREG 09333) is in progress for a US EPR in the framework of USNRC Design Certification.

EDF and Areva, who are co-applicants for Generic Design Acceptance for the UK PWR, remain actively aware of international developments in reactor design, operation and regulation through participation in a range of international organisations. In particular EDF is a member and active participant in the World Association of Nuclear Operators and Areva chairs the Framatome Reactor Owners Group.

7. COMPARISON WITH UK SIZEWELL B DESIGN

The EPR design is the latest development of the PWR design process in Europe. It incorporates design concepts and features from a number of plant designs including the N4 and Konvoi plants. To support the pre-licensing process in the UK, the design enhancements introduced by the EPR design, have been briefly compared with design features of the UK PWR operating at Sizewell 'B'. Whilst the EPR is not a direct evolution of the Sizewell 'B' design, it is considered appropriate to compare the most important improvements in the EPR design to a plant of the same basic technology that has been licensed for the UK. A number of the most significant examples of these developments are discussed below.

The EPR is a plant of significantly higher capacity than the Sizewell 'B' plant, with the rated thermal power being increased to 4,500 MW compared to 3,425 MW. This increased power has been accommodated by an increase in the number of fuel assemblies to 241 compared to 193 in the Sizewell 'B' design. In addition, the active core height is 420 cm compared to a value of 366 cm at Sizewell 'B'. These increases have resulted in a reduction in the core average heat flux, allowing higher local peaking factor to be accommodated without eroding the margins to plant safety. This leads to increased flexibility in core designs and higher fuel utilisation, reducing the amount of radioactive waste created per unit of electricity generated.

The plant has been designed for an operating life of 60 years. This is a significant increase over the Sizewell 'B' design life was initially set at 40 years. Whilst consideration is being given to extending the design life of the Sizewell 'B' plant to 60 years, the scope will remain for EPR to undertake a similar exercise, when actual plant operating data is available, and similarly extend the operating period. The increased design life of the station will decrease the amount of radwaste generated by the plant when measured against the energy generation during the full plant life cycle.

The number of control rods on the plant has been increased to above the number required to maintain a shutdown margin consistent with that adopted on other plants, increasing the margins to re-criticality following shutdown.

In addition to increases in size to reflect the higher thermal capacity of the plant, a number of design improvements have been incorporated. The volumes of the water and steam in the pressuriser have been increased beyond that required to accommodate the power increase. This reduces the impact of transients on the plant pressure and reduces the likelihood of demands being placed on the primary relief valves. Combined with an increase in the volume of water above the core in the reactor pressure vessel, this increases the time to core uncovery following small LOCA transients.

Similarly, the volume of the secondary side of the steam generators has been increased beyond that required for the increased power, increasing the mass of water during normal operation. The ability of the plant to remove decay heat following reactor trip is improved to the extent that the operator has more than 30 minutes to re-establish a heat sink following the loss of all feed. This is a significant improvement relative to the Sizewell 'B' design where operator action is required in less than 30 minutes for the more severe transients.

A further enhancement to the overall plant design has been the change to the in core flux mapping instrumentation. The design concept introduced by the Konvoi design, of instruments inserted from the top of the core, has been adopted. This has removed the requirement for multiple penetrations through the reactor vessel lower head which is now of a single forging design. This significantly reduces the potential for loss of cooling accidents below the elevation of the core.

The refuelling water storage tank (IRWST) is located inside the containment, removing the need for extensive valve changeovers for recirculation for large LOCA inside containment. The four independent train design of the safety injection system allows the incorporation of four sumps within the IRWST, each supporting the operation of one MHSI and one LHSI pump, trained together with an accumulator to deliver to a single RCS loop. The removal of the requirement for containment spray in design basis faults, consistent with the latest Sizewell 'B' approach, allows all four LHSI pumps to be utilised for injection and residual heat removal duties.

The shutoff head of the MHSI has been reduced relative to the Sizewell 'B' design. By utilising a shutoff head below the secondary relief valve setpoint, the potential for SG overfill in SGTR has been reduced. This significantly reduces the likelihood of high levels of activity being discharged to the environment following SGTR.

The design for the ESWS/CCWS is consistent with the four independent trains of safeguards and retains full independence of the individual trains, a significant enhancement relative to the Sizewell design where only two trains of ESWS are provided, each cooling two CCWS trains. This increase in diversity and redundancy is replicated throughout the safety systems design with four 100% capacity safeguards trains provided. In addition train separation is significantly enhanced by the provision of four separate safeguards buildings arranged around the reactor building to reduce the possibility that all four could be lost following a significant external hazard.

To increase the retention of radioactivity within the reactor building following a severe accident with melt through of the reactor pressure vessel, a core melt retention system is provided. A sacrificial cell is incorporated under the reactor pressure vessel to cope with multiphase releases. Once all materials have been captured, burn through of a release gate allows flow of the material to its final destination where it can spread and be cooled. This leads to the safe enclosure of the debris, removing the requirement for further thermo-chemical constraints on the melted material.

Enhanced protection for external hazards is provided, with the reactor building and mechanical safeguard buildings being protected from aircraft crash by design. This is achieved by decoupling the containment interior structures from the outer walls. Physical separation of the remaining components of the safeguards systems ensures the survivability of sufficient equipment to ensure attainment of a safe shutdown state following an aircraft crash hazard.

The EPR design includes a "Safety and Information Control System" which is based on hardwired technology - it includes post-accident monitoring and manual controls for safety system functions. This ensures the operator can shutdown and control the plant in the event that failures occur in the main protection and control systems.

8. COMPARISON WITH INTERNATIONAL SAFETY STANDARDS

8.1. ASSESSMENT AGAINST WENRA REFERENCE LEVELS

EDF has conducted a detailed compliance analysis between the existing EDF nuclear power plants with the WENRA Reference levels (RLs), (January 2007 version), which has been shared with the French Regulator. Only a small number of reference levels are not yet implemented on the French fleet of reactors, covering staff justifications issues, content of the Periodic Safety Reviews and certain PSA applications that the French Regulator was, up to now, reluctant to accept. Discrepancies with one design related RL dealing with the single failure criteria (E 8.2) was also identified, but the wording of this RL is still being discussed within WENRA with a view to either modifying it or finding an acceptable interpretation.

For the EPR project, at this stage of pre-licensing, it is mainly the design issues covered in E, F, G, N, O and S that have to be considered.

The design basis envelope for the EPR fully complies with Issue E RLs. In particular the list of Plant Initiating Events considered in the design is consistent with the one proposed by WENRA.

RCC-A and RCC-B conditions considered in the EPR design allow full compliance with beyond design basis accidents as well as severe accidents Issue F RLs.

The SSC classification of EPR is in accordance with Issue G RLs.

The content of the EPR Preliminary Safety Analysis Report is fully consistent with issue N RLs.

The only point of non compliance is with the Issue O RLs, where up to now, only a level 1 PSA has been performed for EPR. A PSA level 2 is being developed both for Flamanville 3 and the UK EPR and the results will be provided in the Pre-Construction Safety Report to be submitted at step 3 of the pre-licensing process.

8.2. ASSESSMENT AGAINST IAEA STANDARDS AND GUIDELINES

Work performed by the WENRA organisation (see above) took into account the IAEA standards and guidelines which have been produced in order to establish a common reference basis amongst European regulators. Even though the scope of the WENRA work is narrower than the scope of the IAEA guidelines, the positive outcome of the assessment of the EPR design against the WENRA reference levels indicates a good compliance of the EPR design with the IAEA requirements within this specific range.

8.3. ASSESSMENT AGAINST EUR REQUIREMENTS

A comparison of the EPR design with the requirements of the Revision B of the EURs (European Utility Requirements for LWR nuclear power plants) was performed in 2000 following the Basic Design Optimisation Phase (see Volume 1 Chapter C.3). This showed a good level of compliance. Since that time, both the EPR design and EUR requirements have evolved and a new comparison has recently been launched, planned for completion by the end of 2008. It is noted that the EUR revision C, which is to be used for the updated comparison, was benchmarked against the IAEA safety standards in 2004. Even though the scope of the EUR and IAEA documentation does not completely overlap, the comparison showed a good level of compliance between the two sets of reference documentation

8.4. CONCLUSION OF COMPARISON WITH INTERNATIONAL SAFETY STANDARDS

In conclusion, it can be seen that several comparisons of the EPR design (at different design stages) with international safety standards have been performed or are in progress. The outcome indicates good compliance between the EPR design with current international standards, which is expected to be confirmed by on-going comparisons with the EUR revision C.

Table 7: General overview of the design differences between EPR projects

	Finland Olkiluoto 3	France Flamanville 3	USA US EPR	Comment
Main features				
Core power output	4300 MWth	4500 MWth	4590 MWth	US EPR accommodates warm sites
Turbo generator technology	Siemens	Alstom	open	Technology neutral
Frequency	50 Hz	50 Hz	60 Hz	
Seismic conditions	0.1 g	0.25 g	0.3 g	Region specific
Safeguard building size	Fits to local needs	Fits to local needs	Larger buildings	Enlargement for: (1) access to I&C cabinets from the back (US requirement) and (2) tropicalization (size of heat exchangers)
Fuel pool	Pool divided into two pools	Single pool	Single pool	Specific Finnish request
Main voltage levels	690V / 11kV	690V / 11kV	480 V / 4.1 kV /13.4 kV	
I&C				
Safety I&C	Digital AREVA TXS with hardwired back up	Digital AREVA TXS	Digital AREVA TXS	Specific Finnish requirement
Operational I&C	Siemens TXP S5	Siemens TXP S5	Siemens T 3000	T 3000 replaces TXP S5
I&C for severe accidents	Separate I&C for severe accidents	Included in safety I&C	Included in safety I&C	Specific Finnish requirement
Equipment				
Extra borating system	Two trains + one extra pump	Two trains	Two trains	Specific Finnish request
Accumulator size	55 m3	47 m3	55 m3	Specific rules for sizing, country specific
Insulation material for pipes and components	Mineral wool	Fibre glass	Mineral wool	Specific EDF request

	Finland Olkiluoto 3	**France Flamanville 3**	**USA US EPR**	**Comment**
Fuel pool cooling system	2 trains	Third diversified train	2 trains	2 train system allows maintenance only when heat load is limited (delay after fuel unloading)
Design and Safety principles				
Reference code for: design, manufacturing ISI (Primary and Secondary systems)	RCC-M + YVL ASME 11(ISI)	RCC-M RSEM	ASME	*YVL guide: Finnish requirements* *ISI: In Service Inspection*
Application of Break Preclusion Concept Main Primary System	Specific (pipe whip restraints)	Yes	Yes	Specific Finnish requirement *LBB: Leak Before Break*
Application of Break Preclusion Concept Main Secondary System	Specific (pipe whip restraints)	Yes for main steam lines only	Yes for main steam lines only	Specific Finnish requirement
2A break on main reactor coolant lines	Design Basis Condition	Design verification with realistic approach as defined in the Technical Guidelines	Risk Informed rulemaking under review	
Steam Generator Tube Rupture mitigation	Early SG isolation based on safety classified signal of activity in main steam line	Automatic RCS cool down prior to SG isolation	Automatic RCS cool down prior to SG isolation	Specific STUK requirement *SG : Steam generator* *RCS : Reactor Coolant System*
Containment venting	Containment venting implemented	No venting	No venting	Venting : Specific STUK requirements, not accepted by French SA
Loss of Ultimate Heat Sink	Containment venting to avoid over pressurization	Possibility to assume that cooling is fed by water from the outfall	Possibility to assume that cooling is fed by water from the outfall	Specific STUK requirements

	Finland Olkiluoto 3	**France Flamanville 3**	**USA US EPR**	**Comment**
Miscellaneous				
MMI / Control Room	AREVA design specific but OL3 oriented	EDF design based on design implemented in French 1500 MWe plant	AREVA design based on OL3 with US requirement.	