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SUB-CHAPTER 1.5 – SAFETY ASSESSMENT AND INTERNATIONAL PRACTICE

Sub-chapter 1.5 provides an overview of the design and safety assessment process for the EPR within France, Finland and the USA, together with an overview of comparisons of the EPR design against international safety standards (the Western European Nuclear Regulators' Association (WENRA) reference levels, International Atomic Energy Agency (IAEA) Safety Standards, and the European Utility Requirements for LWR nuclear power plants).

1. SAFETY ASSESSMENT IN FRANCE

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 defined 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:
  - minimising 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, overseen by the French and German Safety Authorities and their technical support organisations. The "Technical Guidelines" [Ref-2] produced at the end of the Post Basic Design Optimisation Phase can be considered as a summary of the outcome of the assessment by these safety bodies.

This process resulted in the development of the design of the Flamanville 3 (FA3) EPR unit. The French Nuclear Safety Authority completed its technical examination of the FA3 Preliminary Safety Report in September 2006, reaching the following conclusions [Ref-1]:

- 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 FA3 design had been subjected to a much broader and thorough examination than previous French reactors at the stage of the Preliminary Safety Report, and that experts from several European countries had contributed to the examination, the French Nuclear Safety Authority delivered a positive opinion on the project.

All the recommendations of the French Nuclear Safety Authority were underwritten by studies performed by its Technical Support Organisation, IRSN, (Institute for Nuclear Safety and Radiation protection). Over the nineteen year design period, more than one hundred EPR design assessment reports were issued by IRSN comprising thousands of pages of detailed technical analysis. These reports were tabled at meetings chaired by the French Standing Group on Nuclear Reactors (GPR), an independent advisory body established to support the French Nuclear Safety Authority, consisting of scientists and engineers from France and other European countries and the USA. Conclusions of the GPR with regard to the IRSN recommendations were transmitted to French Nuclear Safety Authority by letter. Section 1.5.1 - Table 1 lists the design assessment reports produced by the IRSN over the review period, which were tabled at meetings with the GPR.

Areas that were subject to the most in-depth regulatory assessment were linked to aspects of the EPR design features that were novel compared with 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 from accident situations (in particular due to hydrogen detonation in case of severe accidents); 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 computerised operation.

The assessment by the French Nuclear Safety Authority concluded with the granting of the FA3 construction license (DAC) in April 2007. The Nuclear Safety Authority then established technical requirements [Ref-3] for the design and construction of FA3.

After the construction license is issued, the next step in the French licensing process is authorisation of first fuel loading. The latter requires a completion of a detailed assessment of the Safety Analysis Report and operating documents, which must be sent to Nuclear Safety Authority at least one year before the first fuel loading.

During the construction period of FA3 technical exchanges with the French Nuclear Safety Authority have continued on the FA3 detailed design, according to a pre-agreed assessment programme time schedule. The aim of the assessment programme schedule is:

- To enable the Nuclear Safety Authority to examine technical features in a timely manner;
- To reduce risk to the construction time schedule to the greatest extent possible.

Under the programme of technical exchanges, an initial version of the start-up authorisation file [Ref-4] was sent to the Nuclear Safety Authority in October 2010. A final version of the file is planned to be sent to the Nuclear Safety Authority in July 2012.

Following the issue of the FA3 construction license (DAC), a control document was issued by the Nuclear Safety Authority containing principles that would be applied for construction surveillance. These covered the following aspects:

- documents to be assessed;
- inspections in design offices and at manufacturing facilities;
- conformity controls;
- inspections on the construction site;
- listing of notification points (witness points) to be mutually defined and agreed.

SECTION 1.5.1 - 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

Report Topic	Date
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 March 2000	May-00

Report Topic	Date
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 computerised 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 computerised operation - Waste zoning - Site related topics Other topics	Jun-06
Radiological consequences of accidents – In operation French plant & EPR	Jun 06
Flamanville 3 EPR - General assessment on Control and Instrumentation system and associated platforms	Jun 09
Radiological consequences of accidents (except severe accident) – In operation French plant & EPR	Jun 09

2. REVIEW OF DESIGN BY FINNISH REGULATOR

2.1. LICENSING PROCESS FOR EPR IN FINLAND

A Construction Licence for the Olkiluoto 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.

2.2. 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. 69 detailed regulatory guides (YVL), produced by STUK, are currently in force.

Assessment of the EPR concept against the YVL guides resulted in modifications being introduced specifically for Olkiluoto 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 RCP [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 presented to the GPR: this did not lead to any recommendation for implementation on FA3. As the EPR proposed for UK is based on the Flamanville 3 EPR design approved by the French Safety Authorities, none of the Olkiluoto 3 modifications listed above are therefore included in the UK EPR design. However, design features specific to Olkiluoto 3 have been considered while reviewing possible modifications to the UK EPR design to confirm that it meets the ALARP principle (see Sub-chapter 17.5).

2.3. 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 finalised and approved in a timely fashion, in parallel with the construction, some delays have occurred to the construction time schedule.

The next licensing milestone is issue of the operating license. 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.

3. SAFETY ASSESSMENT IN THE USA

3.1. INTRODUCTION

AREVA submitted a formal application for design certification of the U.S. EPR to the U.S. Nuclear Regulatory Commission (NRC) on December 11, 2007. The NRC is currently engaged in the review of AREVA's application. This section provides an overview of the NRC's design certification process, describes the history of the pre-application review, and discusses the transition to, and current status of, 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 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 standardised, "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 familiarise 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 completed the pre-application process in December 2007 and subsequently tendered a DC application to the NRC. The NRC conducted an acceptance review of the DC application beginning in January 2008; the application was accepted for review and docketed in February 2008. The NRC is currently proceeding with its comprehensive technical review of the application. Design certification will be completed when the NRC issues a final DC rule for incorporation in the NRC's regulations. The following discussion summarises the history of the pre-application process.

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 2005, as scheduled.

In the January 2006 meeting, 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 14 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 being arranged (several covering multiple topics) and 11 topical or technical reports being produced.

Meetings and submission of additional reports continued through 2007 as Phase 2 of the pre-application process drew to a close and the DC application materials approached completion for submission to the NRC in December 2007. All planned pre-application reports and meetings were completed as of December 6, 2007.

As the NRC's reviews of AREVA's topical reports progressed, AREVA received questions from the NRC, formally called Requests for Additional Information (RAIs). This is a normal part of the review process, whereby the NRC seeks additional details or clarifications with regard to information contained in the report to support the development of the NRC staff's Safety Evaluation Report (SER) on the topical report. Follow-up discussions and meetings with the NRC to discuss RAI responses have been conducted throughout the review process to ensure that AREVA understood the scope of the RAIs and that proposed responses would provide the information needed.

Completion and issue of an SER on a topical report signals that the NRC has found the report to be acceptable from a regulatory standpoint. The information in the topical report may then be referenced in developing the DC application. By the end of 2007, the NRC had issued SERs on five of AREVA's topical reports: the Quality Assurance Plan Topical Report (SER issued 4/26/2007), the Codes and Methods Applicability Topical Report (SER issued 8/8/2007), the Severe Accident Evaluation Topical Report (SER issued 11/29/2007), the Critical Heat Flux (CHF) Correlation Topical Report (SER issued 12/5/2007), and the Instrument Setpoint Methodology Topical Report (SER issued 12/20/2007). Reviews of the other topical reports continued in parallel with the U.S. EPR design certification review, as discussed below.

On October 15-19, 2007, the NRC sent a large team of reviewers to AREVA's Lynchburg offices to conduct a "pre-submission audit" of the documentation that will comprise the design certification application. The objective of the audit was to determine, allowing for additional work to be conducted over the following two months, if the application material would be likely to meet the NRC's "completeness" requirements during the formal acceptance review. The NRC identified a small number of items in the documentation that had to be addressed before the design certification application was submitted, and concluded that if those items were addressed, the application would be ready for the acceptance review. Considering that the NRC reviewed over 10,000 pages of material, AREVA considered the outcome of the audit to be a significant success.

3.4. U.S. EPR DESIGN CERTIFICATION REVIEW

AREVA submitted its DC application on December 11, 2007. The NRC then conducted an acceptance review of approximately 60 days' duration to determine whether the information in the application was sufficient for the agency to initiate its technical review. By letter dated February 25, 2008, the NRC notified AREVA that it had determined that the application met the NRC's acceptance criteria and had been docketed, which is the legal process reflecting its formal acceptance for review. On March 26, 2008, the NRC issued a letter to AREVA establishing its proposed review schedule, comprising the technical assessment of the plant design, culminating in the issue of a Final Safety Evaluation Report (FSER), followed by the formal rulemaking process. Subsequently, the NRC has issued a letter to AREVA on May 21, 2012 updating this review schedule. Issue 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.

The NRC's official schedule projects completion of the technical review in March 2014 and issue of the FSER in July 2014, following its review and endorsement by the Advisory Committee on Reactor Safeguards (ACRS). The rulemaking process is estimated to take most of 2014 to complete, with issue of the final rule in December 2014.

Since submitting the application, AREVA has met frequently with members of the NRC staff on a wide range of technical issues associated with the application. Given the large number of meetings held, the complete list is not detailed here. AREVA has also continued to respond to NRC RAIs on the contents of the DC application and the associated topical reports.

As noted above, the NRC has continued to review AREVA's topical reports in parallel with the DC application. The SER for the Piping Analysis and Pipe Support Design Topical Report was issued on August 8, 2008. Draft SERs have been received for the In-core Trip Setpoint and Transient Methodology Topical Report (August 20, 2009); the U.S. EPR Rod Ejection Accident Methodology Topical Report (July 19, 2010); the Realistic Large Break Loss-of-Coolant Accident Topical Report (August 17, 2010); and the Software Program Manual Topical Report (December 16, 2010). Review continues on the Fuel Assembly Mechanical Analysis Topical Report and In-core Trip Setpoint and Transient Methodology Topic Report, with draft SERs expected the latter part of 2013.

3.5. KEY TECHNICAL ISSUES

The subjects of the topical and technical reports discussed above are representative of the key technical issues on which the NRC is focusing 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. 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 depressurisation capability. The ability of AREVA's safety analysis codes to model the operation of this system has been an important element of the NRC's review. Another area of NRC interest is containment pressure control during design basis accidents without safety-related sprays or fan coolers. The NRC also confirmed 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, as indicated by the approval of AREVA's topical report on this subject

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) modelling 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 NRC's concerns in this area have focused primarily on issues related to communications independence between safety-related and non-safety-related systems.

A key issue that has developed, in part, from the NRC's oversight of operating reactors, is the need to assure adequate long-term cooling after a LOCA. The potential for latent and accident-generated debris and the products of chemical reactions to interfere with recirculatory flow from the containment sump (or in-containment refuelling water storage tank) has become a significant concern. Resolution of this issue for the U.S. EPR by means of a combination of testing and analysis is continuing.

Another technical issue that affects all advanced reactor designs is the need to address the plant's ability to withstand the impact of a large commercial aircraft. In 2009, the NRC promulgated a new regulation requiring an assessment of this hypothetical event. Analysis of the plant response is to be consistent with the approach taken for severe (beyond-design-basis) accidents for U.S. plants. AREVA issued a supplement to the DC application addressing this issue in late 2009.

3.6. 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 structure of the MDEP was reorganised after the previous discussion of the program, and no longer reflects a series of implementation "stages". Rather, design-specific working groups are being established to promote cooperation between participating regulators on plant designs of interest. A parallel effort establishing working groups to look at "harmonising" international approaches on generic technical issues (e.g. codes and standards) has also commenced. The MDEP EPR working group currently includes regulators from France, Finland, the U.K., and the U.S.; other regulators may join the group when and if they decide to do so and meet the MDEP criteria for membership.

3.7. CONCLUSION OF NRC DESIGN REVIEW

Section 3 has summarised the NRC's design certification process, the history 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 in December 2007, and the current status of the NRC's review of the DC application. Over the course of the pre-application review, AREVA conducted an extensive series of meetings with the NRC staff, focusing on significant technical elements of the plant design and associated analytical models, and submitted 23 topical and technical reports on these subjects. The NRC accepted AREVA's application for review in early 2008 and is currently conducting its technical review of AREVA's documentation. Issue of the FSER is scheduled for July 2014, with issue of the final design certification rule expected in December 2014.

4. COMPARISON WITH INTERNATIONAL SAFETY STANDARDS

4.1. INTRODUCTION

EDF has conducted a detailed compliance analysis between the existing EDF nuclear power plants with the WENRA Reference levels (RLs), (January 2007 version) [Ref-1], 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, 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.

RRC-A and RRC-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.

PSA has been performed as an integral part of the EPR design. A comprehensive PSA containing full scope Level 1 and 2, and a UK specific off-site consequences analysis is presented in PCSR Chapter 15 (including an analysis of internal and external hazards and a Seismic Margin Assessment). The off-site PSA model has been used for the UK EPR in support of the ALARP assessment. Chapter 17 of the PCSR provides an ALARP analysis and a review of the PSA results compared to the SAPs numerical targets for risk. The characteristics of the UK EPR PSA allow full compliance with PSA Issue O. (Note: compliance with Issue O dealing with the use of PSA during operation of the plant is not addressed as it is outside the scope of GDA).

Some aspects of the fire protection included in Issue S are the responsibility of the duty holder and are outside the scope of GDA. For all other fire protection issues S, compliance is deemed to have been achieved.

4.2. 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.

Moreover, an evaluation of the UK EPR design was recently conducted by the IAEA. This was done using a selected set of IAEA Safety Standards, having as the principal basis for evaluation, the IAEA Draft Safety Assessment Requirements.

Based on this review, it appeared that the EPR design conforms to the applicable IAEA Fundamental Safety Principles. No fundamental safety problems were identified though it is recognised that a certain number of areas (particularly those presenting novel features) would require additional assessment.

4.3. ASSESSMENT AGAINST EUR REQUIREMENTS

A comparison of the standard EPR design with the requirements of the Revision B of the EURs (European Utility Requirements for LWR nuclear power plants) [Ref-1] was performed in 2000 following the Basic Design Optimisation Phase. This showed a good level of compliance. Since that time, both the EPR design and EUR requirements have evolved and a new comparison was completed in 2009 [Ref-2]. This confirmed that the EPR complied with almost all of the EURs and in some cases exceeded them. It is noted that the EUR Revision C [Ref-3], which was 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 consistency between the two sets of reference documentation

4.4. 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 and current international standards, which is expected to be confirmed by on-going comparisons with the EUR Revision C.

SUB-CHAPTER 1.5 – REFERENCES

External references are identified within this sub-chapter by the text [Ref-1], [Ref-2], etc at the appropriate point within the sub-chapter. These references are listed here under the heading of the section or sub-section in which they are quoted.

1. SAFETY ASSESSMENT IN FRANCE

[Ref-1] Statement concerning the draft decree authorising the creation of the basic Flamanville 3 nuclear installation, including an EPR type nuclear reactor, on the Flamanville (Channel) site. Statement No. 2007-AV-0016. (French) Nuclear Safety Authority, 16 February 2007.

(Avis relatif au projet de décret autorisant la création de l'installation nucléaire de base dénommée Flamanville 3, comportant un réacteur nucléaire de type EPR, sur le site de Flamanville (Manche). Avis n° 2007-AV- 0016. Autorité de Sûreté Nucléaire. 16 février 2007).

[Ref-2] "Technical Guidelines for the design and construction of the next generation of nuclear pressurized water plant units" adopted during plenary meetings of the GPR and German experts on the 19 and 26 October 2000. (French) Nuclear Safety Authority, 20 October 2000. (E)

[Ref-3] Decision with regard to Électricité de France - Société Anonyme (EDF-SA) setting the requirements concerning the nuclear power reactor site at Flamanville (Manche department) for the design and construction of the “Flamanville 3” reactor (BNI 167) and for operation of the “Flamanville 1” (BNI 108) and “Flamanville 2” (BNI 109) reactors. Decision No. 2008-DC-0114. (French) Nuclear Safety Authority, 26 September 2008.

(Décision fixant à Électricité de France – Société Anonyme (EDF-SA) les prescriptions relatives au site électronucléaire de Flamanville (Manche) pour la conception et la construction du réacteur « Flamanville 3 » (INB n°167) et pour l'exploitation des réacteurs « Flamanville 1 » (INB n°108) et « Flamanville 2 » (INB n°109). Décision n°2008-DC-0114. ASN (Autorité de sûreté nucléaire). 26 septembre 2008).

[Ref-4] Robert Pays. EPR FA3 – Transmission of the first working draft version of the start-up authorisation file. Letter ECEP102828. EDF Direction Production Ingénierie. 29 October 2010.

(Robert Pays. EPR FA3 – Transmission de la première version de travail du dossier de demande de mise en service. Lettre ECEP102828. EDF Direction Production Ingénierie. 29 Octobre 2010).

4. COMPARISON WITH INTERNATIONAL SAFETY STANDARDS

4.1. INTRODUCTION

[Ref-1] WENRA Reactor Safety Reference Levels. Western European Nuclear Regulators' Association, Reactor Harmonization Working Group, January 2007. (E)

4.3. ASSESSMENT AGAINST EUR REQUIREMENTS

[Ref-1] European Utility Requirements for LWR nuclear power plants) Volume 1, Chapter C3, Revision B. EUR, Villeurbanne, France. (E)

[Ref-2] European Utility Requirements for LWR nuclear power plants) Volume 3 – Standard EPR Subset. Revision B. EUR, Villeurbanne, France. July 2009. (E)

[Ref-3] European Utility Requirements for LWR nuclear power plants) Volume 1. Revision C. EUR, Villeurbanne, France. April 2001. (E)