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Generic Design Assessment – New Civil Reactor Build

Step 4 Fault Studies – Containment and Severe Accident Assessment of the EDF and AREVA UK EPR™ Reactor

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PREFACE

The Office for Nuclear Regulation (ONR) was created on 1st April 2011 as an Agency of the Health and Safety Executive (HSE). It was formed from HSE's Nuclear Directorate (ND) and has the same role. Any references in this document to the Nuclear Directorate (ND) or the Nuclear Installations Inspectorate (NII) should be taken as references to ONR.

The assessments supporting this report, undertaken as part of our Generic Design Assessment (GDA) process, and the submissions made by EDF and AREVA relating to the UK EPR™ reactor design, were established prior to the events at Fukushima, Japan. Therefore, this report makes no reference to Fukushima in any of its findings or conclusions. However, ONR has raised a GDA Issue which requires EDF and AREVA to demonstrate how they will be taking account of the lessons learnt from the events at Fukushima, including those lessons and recommendations that are identified in the ONR Chief Inspector's interim and final reports. The details of this GDA Issue can be found on the Joint Regulators' new build website www.hse.gov.uk/newreactors and in ONR's Step 4 Cross-cutting Topics Assessment of the EDF and AREVA UK EPR™ reactor.

EXECUTIVE SUMMARY

This report presents the findings of the Fault Studies assessment of the Design Basis Containment Thermal Hydraulics response and Severe Accidents of the UK EPR reactor undertaken as part of Step 4 of the Health and Safety Executive's (HSE) Generic Design Assessment (GDA). My assessment has been carried out on the November 2009 Pre-construction Safety Report (PCSR) and supporting documentation submitted by EDF and AREVA during Step 4.

Only very limited work was performed in the area of Design Basis Containment Thermal Hydraulics and Severe Accidents during Step 2 and 3. The scope of the GDA Step 4 assessment was therefore to review the safety case of the UK EPR reactor in these technical areas and by examining the evidence, supporting arguments and claims made by EDF and AREVA, to make a judgement on the adequacy of the PCSR and its supporting documentation.

It is seldom possible, or necessary, to assess a safety case in its entirety, therefore sampling is used to limit the areas scrutinised, and to improve the overall efficiency of the assessment process. Sampling is done in a focused, targeted and structured manner with a view to revealing any topic-specific, or generic, weaknesses in the safety case. The areas identified for sampling in Step 4 were set-out in advance in an assessment plan based upon the findings of the GDA Step 3 report.

My assessment has focussed on:

- thermal hydraulics challenges to the containment during design basis accident conditions;
- strategy for severe accident progression management;
- key features of the design to mitigate against the consequence of a severe accident;
- challenges to the containment hydrogen control and management system; and
- aspects of validation of the computer codes employed to support the claims within the safety submissions.

It is implicit in the judgements made in the transient analysis, specifically in relation to those faults which subject the containment to thermal and pressure loads, that the containment remains intact when those loads are within the design basis. It is necessary to check, therefore, that the safety case adequately demonstrates that accidents claimed to be within the design basis do not subject the containment to loads which might cause its failure. It is also necessary to ensure that the codes used in the analysis do predict the loads with a high level of numerical confidence on the containment when subjected to these Design Basis Accident (DBA) conditions. It should be noted that the structural behaviour of the containment in response to these calculated loads is reviewed within the civil structural assessment area and is reported separately.

A severe accident commences when loss of emergency core cooling functions have failed to maintain the core in a coolable geometry and, importantly, the core fails to remain in a stable configuration. In order to achieve the international consequence targets, the UK EPR has dedicated severe accident mitigation measures such as the primary depressurisation system and Core Melt Stabilisation System (CMSS) that are 'novel' compared with the existing Pressurised Water Reactors (PWR).

The CMSS is intended to control the core melt progression phase to the movement of molten core debris into the Reactor Pressure Vessel (RPV) lower head, and the subsequent release of core debris into the reactor pit for conditioning, and melt progression through the transfer channel and final melt stabilisation within the core spreading compartment. I have examined the various key features of the evolved design, which are largely based on experimental insights together with the supporting computational analysis.

It has been agreed with EDF and AREVA that it is more appropriate to assess the proposed Technical Specifications, Emergency Operating Procedures (EOP) and the Operating Strategies for Severe Accident (OSSA) management, and the site-specific radiological consequence assessments during the site licensing process. Hence, these items are considered as being outside the scope of the GDA process and have not been included in my assessment.

The summary of my assessment is given in this report with highlights below:

- The UK EPR safety submissions claim that the plant containment design can withstand the various thermal hydraulics challenges in Design Basis Accident (DBA) conditions; together with a hydrogen management control system to minimise the challenges to containment integrity during a severe accident. I have examined the effective performance of the containment cooling system and the CONVECT system, that is intended to bring the inner and outer containment into a single enlarged volume, and the supporting analysis. I have made a number of observations in my assessment, and I judge that these can be resolved during the licensing activities and have, therefore, concluded that an adequate safety case has been provided in this area.
- I have examined the thermal hydraulic pressure and temperature calculations presented in the PCSR from the point of view of the physical processes described in the codes. This examination established that the conservatisms in the assumed parameters and correlations have been identified, to ensure the bounding fault conditions had been analysed, which therefore, established the acceptability of the design pressure and temperatures. These analyses demonstrate that the maximum design pressure and temperatures do not exceed the design limits. Particularly significant is the use of condensation heat transfer correlations in the passive heat sink models, and the claims that these correlations are conservative. In my judgement an acceptable case has been made to support the containment thermal hydraulics response in design basis accidents.
- I have also, in conjunction with the Chemistry Topic Area, examined the extent of the hydrogen release during severe accidents scenarios and the performance of the hydrogen control scheme using Passive Autocatalytic Recombiners, (PAR). The functioning and performance of these PARs is essential to successful management of hydrogen control within specified limits during a severe accident. The PAR’s performance needs to be assured from contaminants and fission products released during accidents. Current research shows that various contaminants, fission product poisoning, and dust are unlikely to adversely impact on the performance of the equipment. However, future confirmatory research and operational feed back will be required to provide additional confidence in efficient performance of these PARs. EDF and AREVA have informed HSE Nuclear Directorate (ND) that they are committed to continue research on certain aspects of the PARs performance in the environment likely to be experienced during accident conditions.
- There are complex phenomena associated with the thermal hydraulics and chemistry linked to the core melt progression and its configuration, transfer and final stabilisation during accident transients. Hence, large uncertainties are associated with predicting this behaviour using the current computer codes. Based on my assessment, Technical Support Contractor (TSC) advice and the results of the independent confirmatory analyses, I can conclude that the safety submissions provide a reasonable understanding of the complex melt phenomena and related interactive processes during transient progression to the core spreading compartment and eventual stabilisation.

During the GDA Step 4 assessment of the UK EPR reactor, I have made a number of observations relating to the shortfalls of evidence in the supporting arguments in the areas of DBA containment thermal hydraulics, severe accident management including hydrogen control and management. EDF and AREVA have responded through the technical discussions and by provision of additional information from their research programmes and computational analysis. This was performed in support of the justification for the claims presented in the safety submission. I expect the revised PCSR will capture the improvements in these areas.

In a number of areas, international research is continuing to further improve the understanding of the phenomena of core melt progression, its composition characteristics within the lower head, and Molten Core Concrete Interaction (MCCI) during the processes associated with CMSS. The research is linked to international initiatives to improve the validated code predictive capabilities in an effort to reduce the uncertainties associated with modelling, and capturing the complex phenomena associated with severe accidents. EDF and AREVA have been active in performing research and development in support of these areas. I have thus encouraged a future Nuclear Site Licensee (NSL) to work with the EDF and AREVA to continue to support these initiatives in order to be an intelligent customer on this important topic.

Although ND will need to assess the additional information that becomes available as the GDA Design Reference is supplemented with additional details on a site by site basis, my judgement is that:

- From my assessment and the results provided by the independent confirmatory analysis, I have concluded that an acceptable safety case has been made for the design features of the UK EPR at the level of details required at PCSR. However, further work is required to improve the confidence in the analysis as these details are progressed during the site licensing activities.
- There are some areas where ND will require additional information to underpin my conclusion, and these are identified as Assessment Findings to be carried forward as normal regulatory business by a future Nuclear Site Licensee for a UK EPR. These are discussed within the report and listed in Annex 1.

Overall, based on the sample undertaken in accordance with ND procedures, I am broadly satisfied that the claims, arguments and evidence presented to support the containment thermal hydraulics response and severe accidents analysis within the PCSR and supporting documentation submitted as part of the GDA process, presents an adequate safety case for the generic UK EPR reactor design. I consider that from a containment thermal hydraulics and severe accidents point of view, the UK EPR reactor is suitable for construction in the UK, subject to assessment of additional information that becomes available as the GDA Design Reference is supplemented with additional details on a site-by-site basis.

LIST OF ABBREVIATIONS

AICC	Adiabatic Isochoric Complete Combustion
ALARP	As Low As Reasonably Practicable
ANL	Argonne National laboratory
ASN	Autorité de Sûreté Nucléaire (French nuclear safety authority)
BDBA	Beyond Design Basis Accident
BMS	(Nuclear Directorate) Business Management System
BSL	Basic Safety level (in SAPs)
BSO	Basic Safety Objective (in SAPs)
C&I	Control and Instrumentation
CCWS	Component Cooling Water System
CDF	Core Damage Frequency
CFD	Computational Fluid Dynamics
CGCS	Combustible Gas Control System
CHF	Critical Heat Flux
CHRS	Containment Heat Removal System
CMSS	Core Melt Stabilisation System
CRDM	Control Rod Drive Mechanism
CMSS	Core Melt Stabilisation System
DBA	Design Basis Accident
DDT	Deflagration to Detonation Transition
DECC	Department of Energy and Climate Change
DfT	Department for Transport
ECCS	Emergency Core Cooling System
EDF and AREVA	Electricité de France SA and AREVA NP SAS
EOP	Emergency Operating Principles
ESWS	Essential Service-Water System
F&B	Feed and Bleed
GDA	Generic Design Assessment
HPME	High Pressure Melt Ejection
HSE	The Health and Safety Executive
HSL	The Health and Safety Laboratory
IAEA	The International Atomic Energy Agency

LIST OF ABBREVIATIONS

IVR	In-Vessel Retention
IRWST	In-Containment Refuelling Water Storage Tank
LOCA	Loss of Coolant Accident
LOFW	Loss of Feed Water
LOOP	Loss of Off-Site Power
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Programme
MCCI	Molten Core Concrete Interaction
MDEP	Multi-national Design Evaluation Programme
MSLB	Main Steam Line Break
NCB	Non Classified Building
ND	The (HSE) Nuclear Directorate
NDA	Nuclear Decommissioning Authority
NSL	Nuclear Site Licensee
OCNS	Office for Civil Nuclear Security
OJEU	Official Journal of the European Union
OSSA	Operating Strategies for Severe Accident
PAR	Passive Autocatalytic Recombiners
PCER	Pre-construction Environment Report
PCMSR	Pre-Active Commissioning Safety Report
PCSR	Pre-construction Safety Report
PDS	Primary Depressurisation System
PID	Project Initiation Document
POR	Passive Outflow Reducer
PRT	Pressuriser Relief Tank
PSA	Probabilistic Safety Analysis
PSR	Preliminary Safety Report
PSR Valves	Pressuriser Safety Relief Valves
PWR	Pressurised Water Reactors
RCS	Reactor Coolant System
RGP	Relevant Good Practice
RIA	Regulatory Issue Action
RIT	Royal Institute of Technology
RO	Regulatory Observation

LIST OF ABBREVIATIONS

ROA	Regulatory Observation Action
RPV	Reactor Pressure Vessel
SAP	Safety Assessment Principles
SARNET	European Severe Accident Research Network
SBLOCA	Small Break Loss of Coolant Accident
SFAIRP	So Far As Is Reasonably Practicable
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIS	Safety Injection Systems
SML	Submission Master List
SSC	System, Structure and Component
SSER	Safety, Security and Environmental Report
STUK	Säteilyturvakeskus (The Finish Nuclear Safety Authority)
TAG	(Nuclear Directorate) Technical Assessment Guide
COT	Core Outlet Temperature
TQ	Technical Query
TSC	Technical Support Contractor
US NRC	Nuclear Regulatory Commission (United States of America)

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Figure 3: Schematic View of the Transfer Channel and the Spreading Area

Figure 4: Road Map of the Codes Utilised in Support of the Severe Accident Analysis

1 INTRODUCTION

1 My report presents the findings of the Step 4 Fault Studies - Containment Thermal Hydraulics Response and Severe Accidents assessment of the November 2009 UK EPR reactor Pre-construction Safety Report (PCSR) (Ref. 12) and supporting documentation provided by EDF and AREVA under the Health and Safety Executive's (HSE) Generic Design Assessment (GDA) process. Assessment was undertaken of the PCSR and the supporting evidentiary information derived from the Master Submission List (Ref. 13). My approach was to assess the principal submission, i.e. the PCSR, and then undertake assessment of the relevant documentation sourced from the Master Submission List on a sampling basis in accordance with the requirements of Nuclear Directorate (ND) Business Management System (BMS) procedure AST/001 (Ref. 2 and 3). I have used the Safety Assessment Principles (SAP) (Ref. 4) as the basis for my assessment. Ultimately, the goal of assessment is to reach an independent and informed judgment on the adequacy of a nuclear safety case.

2 During my assessment a number of Technical Queries (TQ) and Regulatory Observations (RO) were issued and the responses provided by EDF and AREVA assessed. Where relevant, detailed design information from specific projects for this reactor type has been assessed to build confidence and assist in forming a view as to whether the design intent proposed within the GDA process can be realised.

3 The UK EPR safety submissions claim that the plant containment design can withstand the various thermal hydraulics challenges in Design Basis Accident (DBA) conditions. Additionally a hydrogen management control system is proposed to minimise the challenges to containment integrity during a severe accident and the design also includes features intended to limit the consequences of a core melt.

4 These features have been designed to achieve international consequence targets. In particular, the design intent is to “practically eliminate[1]” a large early release of radiation to the environment in the event of a severe accident, allowing time for mitigation measures to be taken and to reduce the consequences of less severe faults to levels where off-site mitigation measures are not required.

5 In pursuing these design objectives, the UK EPR has included dedicated severe accident mitigation measures that are ‘novel’ to existing Pressurised Water Reactors (PWR). The design intent is welcome and since it goes beyond relevant good practice in the UK, is a step towards the requirement to demonstrate that reasonably practical measures have been taken to address the risk of severe accidents. I also note that these plant design features are based on Olkiluoto 3 (OL3) in Finland and Flamanville 3 (FA3) in France which have been supported by research and development activities. The focus of my assessment has therefore been to satisfy myself that the design intent has been realised in practice.

6 The UK EPR is designed with a large containment building so that active measures to control containment pressure are not claimed immediately following an accident and provision for long term cooling of the containment has been made. These provisions are assessed in Section 4.1.

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[1] The possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if the conditions can be considered with a high degree of confidence to be extremely unlikely to arise (from IAEA NSG1.10).

7 The EPR containment is designed based on a two-region concept; inner containment (inaccessible) and outer containment where limited access to equipment within the containment building is permitted while the reactor is operating at power. This is facilitated by the provision of radiation shielding within the containment and also thin contamination barriers. These provisions comprise the CONVECT system. The CONVECT system is intended to bring the two volumes of inner and outer containment into a single volume during accident conditions.

8 The containment performance in accidents is dependent on the effective performance of the containment cooling system and the CONVECT containment volume enlargement system. My assessment of this system is reported in Section 4.1.2.

9 The most notable novel feature of the design is the provision of a facility to contain molten core material in the event of it escaping from the reactor pressure vessel, and onwards into the reactor pit and final melt stabilisation within the core spreading compartment, commonly referred to as the “core catcher”. This is assessed in Section 4.3.

10 There are complex phenomena associated with the thermal hydraulics and chemistry associated with the melt progression and configuration, transfer and final stabilisation during accident transients. Hence, large uncertainties are associated with predicting this behaviour using the current computer codes. Given these complexities, I commissioned a set of independent confirmatory analyses to gain an understanding of the level of uncertainties; details of my confirmatory analyses are presented in Section 4.3. The phenomena of steam explosions and re-criticality are also discussed in Section 4.4.

11 The UK EPR design relies on Passive Auto-Catalytic Recombiners (PAR) to limit the global hydrogen levels within containment. These measures are considered in Section 4.5.

12 Measures to mitigate the effects of the release of fission products from the containment include the provision of a robust double-skinned containment with filtered air extraction from the interspace and a steel liner on the inner skin. The effectiveness of the UK EPR approach is assessed in Section 4.6.

13 The strategy used for my assessment within Step 4 of GDA is outlined below together with the standards against which the safety case has been judged.

2 NUCLEAR DIRECTORATE'S ASSESSMENT STRATEGY FOR FAULT STUDIES - CONTAINMENT THERMAL HYDRAULICS RESPONSE AND SEVERE ACCIDENTS

14 Only very limited work was performed in the area of Design Basis Containment Thermal Hydraulics performance and Severe Accident during GDA Steps 2 and 3. The scope of the GDA Step 4 assessment was therefore to review the safety case of the UK EPR reactor in these technical areas and by examining the claims, supporting arguments and evidence made by EDF and AREVA, to make a judgement on the adequacy of the PCSR and its supporting documentation. The intended assessment strategy for GDA Step 4 was set out in an assessment plan that identified the intended scope of the assessment and the standards and criteria that would be applied. This is summarised below:

2.1 Assessment Plan

15 The plan for assessment of Containment Thermal Hydraulics Response and Severe Accident topic area in GDA Step 4 is set out in Ref. 1.

16 The technical assessment in the Fault Studies - Containment Thermal Hydraulics Response and Severe Accident topic area only commenced part way through the GDA Step 3 process. For this reason, the scope of the assessment only included certain aspects of the severe accident analysis at that stage. I have therefore included those areas that would have been reviewed in GDA Step 3. For example, in GDA Step 4, the scope of the assessment was extended to examine the thermal hydraulic analysis performed in support of the Probabilistic Safety Analysis (PSA) success criteria.

17 Particular focus was placed on the evidence required to support the claimed values for safety limits presented as design criteria in the safety case. My assessment focused on the following topics;

- thermal hydraulics challenges to the containment during design basis accident conditions;
- strategy for severe accident progression management;
- key features of the design which mitigate against the consequence of a severe accident;
- performance of the containment hydrogen control and management system;
- adequacy of the evidence supporting the claims and arguments assessed within GDA Step 3; and
- validation and use of the computer codes employed in relation to containment thermal hydraulics and severe accident to support the claims within the safety submissions.

18 In selected cases, I have commissioned independent confirmatory analyses from Technical Support Contractors (TSC).

19 The specific issues relating to the adequacy of the hydrogen management and control system to minimise the challenges to containment integrity during a severe accident have also been included within the assessment at GDA Step 4.

2.2 Standards and Criteria

20 The standards and criteria that are used to judge the UK EPR are defined in the 2006 HSE Safety Assessment Principles for Nuclear Facilities (SAP) (Ref. 4). These principles require a robust demonstration of the design against conservative design assumptions for postulated faults considered within the design basis. The bulk of the assessment principles provide guidance for the assessment of these faults.

21 In the case of very low frequency events which potentially lead to severe accidents, a different set of requirements apply. These requirements are designed to require a demonstration that measures have been taken to mitigate the risk associated with the faults to a level that is As Low As Reasonably Practicable (ALARP). In these cases, the assessment is focused on confirming that appropriate mitigation measures have been applied and that the cost of further safety measures would be disproportionate to the potential reductions of risk.

22 The following principles, taken from Ref. 4, are considered relevant to the assessment of the containment thermal hydraulics performance and severe accident have been used:

- **EKP.1: Engineering principles: key principles – Inherent safety**
  The underpinning safety aim for any nuclear facility should be an inherently safe design, consistent with the operational purposes of the facility.

- **EKP.2: Engineering principles: key principles – Fault tolerance**
  The sensitivity of the facility to potential faults should be minimised.

- **EKP.3: Engineering principles: key principles – Defence in depth**
  A nuclear facility should be so designed and operated that defence in depth against potentially significant faults or failures is achieved by the provision of several levels of protection.

- **ECS.4: Engineering principles: safety classification and standards – Codes and standards**
  For structures, systems and components that are important to safety, for which there are no appropriate established codes or standards, an approach derived from existing codes or standards for similar equipment, in applications with similar safety significance, may be applied.

- **ECS.5: Engineering principles: safety classification and standards – Use of experience, tests or analysis**
  In the absence of applicable or relevant codes and standards, the results of experience, tests, analysis, or a combination thereof, should be applied to demonstrate that the item will perform its safety function(s) to a level commensurate with its classification.

- **EDR.4: Engineering principles: design for reliability – Single failure criterion**
  During any normally permissible state of plant availability no single random failure, assumed to occur anywhere within the systems provided to secure a safety function, should prevent the performance of that safety function.

- **FA.1: Fault analysis: general – Design basis analysis, PSA and severe accident analysis**
  Fault analysis should be carried out comprising design basis analysis, suitable and sufficient PSA, and suitable and sufficient severe accident analysis.

- **FA.2: Fault analysis: general – Identification of initiation faults**
  Fault analysis should identify all initiating faults having the potential to lead to any person receiving a significant dose of radiation, or to a significant quantity of radioactive material escaping from its designated place of residence or confinement.

- **FA.3: Fault analysis: general – Fault sequences**
  Fault sequences should be developed from the initiating faults and their potential consequences analysed.

- **FA.4: Fault analysis: general – Fault tolerance**
  DBA should be carried out to provide a robust demonstration of the fault tolerance of the engineering design and the effectiveness of the safety measures.

- **FA.9: Fault analysis: general – Further use of DBA**
  DBA should provide an input into the safety classification and the engineering requirements for systems, structures and components performing a safety function; the limits and conditions for safe operation; and the identification of requirements for operator actions.

- **FA.15: Fault analysis: severe accident analysis – Fault sequences**
  Fault sequences beyond the design basis that have the potential to lead to a severe accident should be analysed.

- **FA.16: Fault analysis: severe accident analysis – Uses of severe accident analysis**
  The severe accident analysis should be used in the consideration of further risk-reducing measures.

- **FA.17: Fault analysis: assurance of validity of data and models – Theoretical models**
  Theoretical models should adequately represent the facility and site.

- **FA.18: Fault analysis: assurance of validity of data and models – Calculation models**
  Calculational methods used for the analyses should adequately represent the physical and chemical processes taking place.

- **FA.19: Fault analysis: assurance of validity of data and models – Use of data**
  The data used in the analysis of safety-related aspects of plant performance should be shown to be valid for the circumstances by reference to established physical data, experiment or other appropriate means.

- **FA.20: Fault analysis: assurance of validity of data and models – Computer models**
  Computer models and datasets used in support of the analysis should be developed, maintained and applied in accordance with appropriate quality assurance procedures.

- **SC.4: The regulatory assessment of safety cases – Safety case characteristics**
  In addition, Paragraph 93 of SC.4: requires demonstration that ALARP has been achieved for new facilities, modifications or periodic safety reviews, the safety case

should:

i) identify and document all the options considered,

ii) provide evidence of the criteria used in decision making or option selection, and

iii) support comparison of costs and benefits where quantified claims of gross disproportion have been made.

The above principles are listed in Table 1.

23 The safety principles listed above are UK specific, but ND also expects that the means of mitigation of the consequences of a severe accident shall also comply with the safety objective number O3 relative to accidents with core melt of the WENRA Statement on safety objectives for new Nuclear Power Plants (NPP) (Ref. 8).

24 In terms of containment and severe accidents the EPR design intent was based on the French and German Utility Technical Guidelines (Ref. 21) for future PWR plant. These Guidelines appear to demand significant improvements at the design stage with respect to consideration and management of severe accident. These guidelines include reduction of Core Damage Frequency (CDF) and accident situations related to the early release of source terms to the public as being practically eliminated. These expectations have been addressed by EDF and AREVA.

2.3 Assessment Scope

25 For the purposes of GDA, the assessment has concentrated on examining the containment thermal hydraulics response in accident conditions and the performance of the systems designed to provide mitigation against the consequences of a severe accident. The specific topics sampled have been based on the findings of the GDA Step 3 Assessment.

2.3.1 Findings from GDA Step 3

26 The Step 3 report identified a number of specific issues which needed addressing by EDF and AREVA in sufficient time to be assessed in GDA Step 4:

- The rationale for the strategy for manual operation of the Primary Depressurisation System (PDS) when severe accident conditions are detected.
- The arguments supporting the creep rupture and possible locations of the weakest point of the primary circuit and any related risk of containment by-pass by Steam Generator (SG) tube failure.
- The basis of the analysis and the validation of the codes used to determine the hydrogen transport and distribution within the containment environment during a core damage event.
- Consideration of the common-mode failure of the hydrogen removal capability of the PARs distributed within the containment.
- Examination of the effects of uncertainties in the transient progression of the molten debris from the core region to its arrival within the core catcher.
- The need for passive and diverse means of venting the containment during fault conditions.

27 In each of these areas, EDF and AREVA have made substantial progress within GDA Step 4 and the detailed findings of my assessment are discussed in Section 4 of this report.

28 I also judge that hydrogen release during degradation of the reactor core, its distribution and flow characteristics within the containment is sufficiently important that I have included the examination of the Combustible Gas Control System (CGCS) generally in my assessment.

2.3.2 Use of Technical Support Contractors

29 Technical Support Contractors (TSC) have been used in a number of areas:

- The development of an independent computer model of the EPR primary circuit, the various mitigation measures and containment systems including detailed reactor core and the proposed core catcher concept to examine all aspects of the transient core melt and the subsequent containment challenges during the severe accident.
- Confirmatory analysis using an independently developed numerical methodology to examine the spreading effectiveness of corium flow within the spreading compartment in severe accident conditions.
- A review of the severe accident mitigation measures - the Core Melt Stabilisation System (CMSS).
- A review of the use of computer codes to model International Standard Problem (ISP) verification studies to provide knowledge and insights on containment hydrogen mixing phenomena.
- A review of the containment thermal hydraulics performance in accident conditions and relevant international good practice.
- The topic of steam explosion phenomena relating to In-Vessel, Reactor Pit and Spreading Compartment has also been examined in a brief review.

30 The contractor review, supported by the confirmatory analysis of the severe accident progression simulating the core melt and degradation, relocation, RPV failure, MCCI and core melt stabilisation, was performed to provide independent verification and confirmation of the claims made within the UK EPR PCSR. This work is reported in Ref. 22 and has provided additional assurance of the timing and severity of key events and consequences of a severe accident. The result of this independent confirmatory analyses work is further described in Section 4.3.4.

31 The UK EPR design includes a CMSS (reactor pit/transfer channel/core spreading area) - the “core catcher”. The design intends to retain and cool the material from the reactor vessel in the event of molten fuel release. This is to prevent pressurisation of the containment building as a result of molten-core concrete interaction and to ensure that the containment base mat is not breached - potentially permitting fission product release into the underlying ground water.

32 The CMSS includes a compartment dedicated to maintaining a stable melt configuration and to establish its longer term cooling. I commissioned an independent confirmatory analysis to examine its behaviour and how efficiently it distributes the molten material within the core spreading area. The result of this confirmatory analysis and the supporting review is reported in Ref. 23 and 24 respectively.

33 The analysis of chemistry and chemical reactions during a severe accident and the status of the composition of debris (Ref. 25) within the lower head have been assessed by my chemistry colleagues. The results of this assessment are reported in (Ref. 26).

34 Similarly, the assessment of chemical behaviour and the performance of the PARs within the environment likely to exist within the containment as a result of a severe accident has been jointly managed with my chemistry colleagues and is reported in Ref. 27. This reference provides some independent confirmation of the claims made.

35 I also commissioned a short review of the international research to examine and consider the relevance of the current knowledge to the areas of the UK EPR design where the risk of steam explosion may exist.

2.3.3 Cross-cutting Topics

36 The following Cross-cutting Topics have been considered within this report:

37 The core fuel melt including all core materials and their interactions and behavioural characteristics during severe accidents has required collaboration. My colleagues in the chemistry topic assessed the chemistry of molten material and chemical reactions during the transient, and I have assessed the issues relating to thermal hydraulics and complex heat transfer processes within and from the melt progression and stabilisation. The assessment included the issues arising from melt chemistry and hydrogen gas released into the containment. This is likely to affect the performance and the subsequent demands within the containment design limits.

38 The performance of individual PARs located within the containment is principally a chemistry issue, but the localised hydrogen distribution arises from conditions prevailing within the containment. It is important that hydrogen within sub-compartments is mixed with the global flow patterns. The PARs performance is also affected by containment sprays, dust, contaminants, and fission products. The impact on PAR performance of these appears to be a short delay in the PAR start-up characteristics which I consider will not adversely affect the containment's performance. The fission product released during a severe accident could potentially change their character from aerosol to gaseous in the high temperature environment of the PAR plates. The EPR design requires that the PARs will be checked for deterioration in performance at maintenance intervals.

39 I have collaborated with my chemistry colleagues in this chemistry area. They have carried out a thorough assessment of this technically challenging area. My concern has been to ensure that there is adequate performance of this mitigation measure necessary to ensure that hydrogen concentration within the containment will not exceed the maximum concentration limits imposed for the containment. The performance of these PARs will significantly influence the hydrogen transport and distribution within the containment together with the design features creating an enlarged containment volume. The issues relating to the assessment of hydrogen transport within the containment have been covered in Section 4.5.

2.3.4 Integration with other Assessment Topics

40 The interaction with other assessment disciplines such as fault studies, chemistry and PSA has inevitably been routine and the three assessment areas have been very closely integrated, with contact on a daily basis. My particular concern has been to ensure that the assumptions on Design Basis Analysis (DBA) and Beyond Design Basis Accident (BDBA) scenarios made in fault studies are considered, and appropriately

assessed for their impact on containment thermal hydraulics and severe accident demands.

41 In performing the confirmatory analyses to examine the plant's performance in severe accident conditions close collaboration was developed with the PSA team to ensure the bounding cases were included in the analyses matrix. The selection of these scenarios was informed by the insights of the PSA discipline and the supporting TSC modelling expertise to provide confidence in the EDF and AREVA submissions.

2.3.5 Out of Scope Items

42 It has been agreed with EDF and AREVA that it is more appropriate to assess the proposed Technical Specifications, Emergency Operating Procedures (EOP) and the Operating Strategies for Severe Accident (OSSA) management and the site-specific radiological consequence assessments during the site licensing process. Hence, these items are considered outside the scope of the GDA process and have not been included in my assessment. But these are noted to be critical to the successful management of a severe accident.

3 EDF AND AREVA'S SAFETY CASE

43 The containment building is provided with a metal liner covering the inside of the pre-stressed concrete inner shell to ensure there are very low leakage rates from inside of the containment building to the external environment. The containment building is double walled; the annulus between the inner and outer shell is kept at a slight negative pressure to enable collection and filtration of any leakage before release to atmosphere from the primary containment. All penetrations emerge into connected buildings so that leakages may be collected and filtered.

44 The plant containment design can withstand the various thermal hydraulics challenges arising in DBA conditions; together with a hydrogen management control system to minimise the challenges to containment integrity during severe accidents. This is dependent on the effective performance of the containment cooling system and the containment atmosphere mixing strategy facilitated by the "CONVECT" system. This system has a high dependency impact on the overall containment performance and the hydrogen mitigation scheme. The CONVECT system is intended to bring the two regions of inner (inaccessible) and outer containment into a single volume during accident conditions.

45 The ultimate heat sink, which is provided by the Essential Service-Water System (ESWS) and the Component Cooling Water System (CCWS), is organised into four separate and independent trains each fitted with a pump and a heat exchanger. In addition, EDF and AREVA claim that this main system is backed up by a dedicated circuit comprising two trains fed by specific power supplies which enables heat from corium cooling to be removed in severe accident conditions in the event of a total loss of heat sink.

46 The UK EPR plant design has four primary circuit loops with emergency cooling systems providing the diversity and redundancy to ensure the risks of moving from design basis accidents to severe accidents are claimed to be very low. The design has also included new mitigation systems to help with the management of severe accidents to ensure any core melt scenarios, which are claimed to be very low frequency accidents, are managed and the plant demands can be ameliorated. The mitigating features in severe accidents are summarised below.

47 The CMSS mitigation system is designed to protect the plant against the consequences of core melt accidents and confines the resulting radioactivity to the containment. It is assisted in the approach through the depressurisation system which lowers the primary pressure and reduces the potential for plant structural failures such as Steam Generator Tube Rupture (SGTR). It is intended to control, manage and spread corium resulting from core meltdown at low pressure to the reactor pit.

48 This system has a transfer channel which directs the gravitational flow of corium from the reactor pit into a large spreading compartment whose floor is covered with a layer of sacrificial material over a network of cooling channels that protects the foundation raft. The thickness of the raft has been increased compared to that of the older plants, thereby preventing penetration by corium. The arrival of the melt in the spreading compartment triggers devices that initiate the gravity driven flow of water from the In-Containment Refuelling Water Storage Tank (IRWST) into the spreading compartment.

49 The UK EPR design incorporates a Containment Heat Removal System (CHRS) which includes a dedicated spray system with heat exchangers and dedicated heat sink to control the pressure rise inside the containment. In severe accident conditions, although the initiating set point is the containment pressure, the spray system can be activated by

operators within 12 hours after entry into severe accident conditions. Besides assisting to limit containment pressures and temperatures the spray system helps to wash fission products into the IRWST where decontamination may occur at high pH. The second mode "active cooling" of operation of the containment heat removal system enables the water to flow directly into the spreading compartment via the other available line or instead of the spray system. The introduction of sub-cooled water over the corium is intended to provide cooling and leads to a reduction of steaming production.

50 The above flow is important to cool the debris, but it is strictly controlled in accordance with the overall severe accident progression management, and to minimise the relevant hazard of steam explosion ex-vessel. EDF and AREVA recognise that there are uncertainties, and the phenomena governing occurrence of steam explosion are complex. EDF and AREVA have therefore made probabilistic arguments and claim evidence from experimental research and development work supports their assertion that in-vessel or ex-vessel steam explosion have a very low probability of occurrence.

51 In case of a severe accident, hydrogen is expected to be produced and released in the three phases of the accident; covering in-vessel core degradation and relocation, ex-vessel phases within the reactor pit and the spreading room. In addition, these three phases will release large quantities of non-combustible gases inside the containment. The pre-stressed concrete inner shell has been designed to withstand the pressure and temperature demands that will result from the combustion of the hydrogen released.

52 Furthermore, the UK EPR plant includes a Combustible Gas Control System (CGCS) that controls and manages the hydrogen combustion inside the containment following transients that develop into severe accidents by maintaining the 'local' atmospheric concentration of hydrogen everywhere in containment to be within prescribed limits. EDF and AREVA propose to install PARs at strategic locations in the containment sub-compartments, dome region and elsewhere, to keep the average hydrogen concentration below 10% at all times to avoid any risk of detonation. EDF and AREVA have given consideration to the proximity of safety systems, cables and walls in relation to selecting PAR locations.

4 GDA STEP 4 NUCLEAR DIRECTORATE ASSESSMENT FOR FAULT STUDIES - CONTAINMENT THERMAL HYDRAULICS AND SEVERE ACCIDENTS

53 The assessment of the UK EPR within GDA Step 4 process, has concentrated on examining the containment thermal hydraulics response in accident conditions and the performance of the systems designed to provide mitigation against the consequences of a severe accident. Whilst there are overlaps between the claims and substantiation provided for all equipment and safety features that are utilised to mitigate against the consequences of accidents presented within the PCSR and the supporting documents, I have reported the findings of my assessment under four main headings;

- containment thermal hydraulics response;
- effectiveness of the measures to depressurise reactor coolant system;
- severe accidents progression with failure to restore cooling; and
- severe accident consequences.

54 The report therefore includes the assessment findings of the containment thermal hydraulics performance and severe accidents.

55 I have also assessed a number of other topics that are closely related to the two topic areas covered above. These are also included within this Section of the report.

4.1 Containment Thermal Hydraulics

56 The UK EPR containment has a double-wall (two thick concrete shells) concept with an annulus and a leak-tight steel liner on the inside surface of the inner-most shell. The inter-space is maintained at a negative pressure to collect any small quantities of leakage from the internal containment volume and to filter it before it may be released into the outside environment via a stack. The 80,000m^3 containment volume is divided into two major sub-regions connected through a system of flaps and dampers intended to separate the accessible (service) area from the inner inaccessible volume, (controlled access).

57 The containment is required to protect the public from any accident state that involves release of radioactivity from the fuel in accordance with the appropriate HSE SAPs. These events are claimed to be very low probability occurrences, but the containment must be a leak-tight barrier against these releases. In order to perform these functional requirements the containment has to be able to accommodate the thermal demands arising from design basis accidents DBA and beyond design basis accidents (BDBA).

58 It is implicit in the judgements made in the transient analysis, specifically in relation to those faults which subject the containment to thermal and pressure loads, that the containment remains intact. It is necessary to check, that the safety case adequately demonstrates accidents claimed to be within the design basis do not subject the containment to loads which might cause its failure, and to ensure that the codes used in the analyses do reasonably predict the load demands on the containment when subjected to these DBA conditions. I assessed the predictive codes used to support the safety justification against the requirements identified in the appropriate HSE SAPs.

59 My assessment addressed those DBA accidents and internal challenges to containment with respect to pressure and temperature limits, penetration seal leakage rates, and adequacy of the containment cooling systems. I examined the passive heat transfer to the containment wall as it is a significant heat transfer route that will influence the

pressure and temperature demands. My assessment included the condensation heat transfer phenomena on walls, structures and components within the containment and other phenomena such as thermal capacity effects that are required to be analysed to make containment performance predictions.

60 The full scope of the UK EPR containment includes the containment environment, the containment isolation system, and the Combustible Gas Control System (CGCS). The containment has also to be able to withstand a number of internal hazards such as fire, and external hazards such as seismic events, flooding and aircraft impact. It should be noted that the structural behaviour of the containment in response to these predicted loads is reviewed within the Civil Structural assessment area and is reported separately, (Ref. 28).

4.1.1 Design Basis Analysis

61 The transformation of the containment from two sub-regions into a single volume is through the concept known as CONVECT whose primary function is to increase the natural convection capability within the containment. This is achieved through the design features located within the sub-compartments in the lower region of the containment. This includes:

- passive pressure sensitive rupture foils above SG compartments,
- passive temperature and pressure sensitive convection foils at the same elevation, and
- mixing dampers that open on both pressure difference and absolute pressure. These are located at the lower parts of the containment to promote natural convection.

62 The system provides the capability to equalise pressure inside the containment during DBA, and is claimed to promote efficient containment mixing to equalise pressure, and to avoid localised accumulation of gases (steam and any non condensable gases) through more efficient mixing following LOCA releases and other releases from the depressurisation system and Pressuriser Safety Relief (PSR) Valves. CONVECT also allows access to increased volumes and surfaces for thermal capacity effects, and thermal hydraulics, heat transfer and condensation effects.

63 I have assessed this concept against the requirements of SAP ECS.4 and FA.4 requiring consideration of engineering principles and effectiveness of safety features to demonstrate that the pressure limits are met in design basis faults and also for the adequacy of mixing when hydrogen is released into the containment. Consideration of the pressure response follows. Hydrogen mixing is discussed in Section 4.5.

4.1.1.1 EDF and AREVA’s Case

64 The controlled access philosophy within the containment considers the operational requirements of the plant during normal operation, shut down conditions, planned outages and accident conditions. Access requirements are part of the general ventilation management system. In response to TQ-EPR-1129 (Ref. 9) relating to the operating philosophy of the ventilation system, EDF and AREVA have provided an overview of the ventilation management in normal operating and accident conditions. The ventilation strategy of the containment has however not been included in the sampling of the PCSR at GDA Step 4 and will therefore be considered during the site specific licensing activities.

The assessment of the ventilation system also generally falls within the assessment remit of other disciplines within HSE ND. (See Ref. 29 for further details)

65 The CONVECT system is employed to promote hydrogen mixing and transportation to the other designed safety system installed for hydrogen management; i.e, 47 passive autocatalytic converters. The PARs are self starting at a hydrogen concentration of approximately 2 vol%. Thus, any DBA steam and non condensable gases in the containment atmosphere, should they occur, will be controlled. The convection dampers are designed to withstand normal pressure and temperature conditions and retain their leak tightness during normal operation or DBA conditions until ambient temperatures exceed 79 to 85°C or a pressure difference of 50 Pa.

66 The containment heat removal system is employed to support the control of pressure inside the containment and to ensure the removal of the requisite decay heat from containment during design basis accidents and severe accidents. The system transfers decay heat energy from the In-Containment Refuelling Water Storage Tank (IRWST) to the ultimate cooling water system. There are various system requirements such as maintaining sufficient suction head to transfer the water to be cooled, and that any line-filters do not get blocked.

67 The containment must be capable of accommodating the temperature and pressure energy demands produced from Loss of Coolant Accidents (LOCA), main steam line breaks (MSLB) and Loss of Off-site Power (LOOP). The design basis used to assess the adequacy of the containment design assumes that the plant thermal power is 4,500 MWth with 2% allowance to be added for uncertainties, and maximum decay heat. A range of DBA accident scenarios are presented to assess the design's capabilities.

68 During the early stages of a LOCA accident scenario as the plant is depressurised core cooling will be maintained through the Medium Head Safety Injection (MHSI) and Low Head Safety Injection (LHSI) systems to use water to control the core temperature rise. Depending upon the size of the leak and its location the time scales of events will vary. Operator action is essential and should identify the leaking loop for isolation to limit the water losses from the RCS. However, if attempts should indicate that core outlet temperatures are rising rapidly then the accident management procedures should provide direction to manage the water losses and inflow of safety coolant into the RCS. The operational requirements of the safety injection post severe accident conditions will be assessed during the site licensing activities when the OSSAs should be made available.

69 EDF and AREVA's computational analysis supports the containment safety justification. The safety submission has examined a range of accident scenarios considered to be within the design basis. These analyses have led to identifying the cases where the energy input from LOCA and steam line breaks is challenging the containment. I have examined the two bounding cases identified within the safety case. The two scenarios are: double ended cold leg break of the reactor cooling system, and the Main Steam Line Break (MSLB) within the containment building. These analyses suggest that the predicted temperatures and pressures do not exceed the design limits with reasonable margins for the worst case scenario of MSLB.

Assessment

70 EDF and AREVA have examined a range of parameters that are considered to be conservative to support the conclusions arising from the bounding analysis. I have examined these assumptions. The modelling uses lumped-parameter computer codes

developed for single-region containments and employs allowances for uncertainty determined as part of the qualification of these codes.

71 I have asked for justification of the applicability of the methods employed to the case of two region containments of the type proposed for UK EPR and have been provided with evidence from separate-effect tests on a limited scale. This evidence supports the assumptions made in the analysis, but given the limited special detail used in the codes employed, uncertainties remain on the scalability of this validation to the EPR containment.

72 Containment analysis computer codes are continuously being improved and are expected to adopt the advances in knowledge and understanding from the wide range of international experiments undertaken on these topics. My expectation is that safety cases and designs should be constructed using verified and validated computer codes to allow the residual uncertainties to be reduced to meet the requirements of SAP FA.18 requiring adequate representation of the physical processes taking place.

73 I am satisfied that, in general, the lumped parameter code for GDA is being used in a way consistent with the current body of evidence, and therefore I conclude that a suitable assessment of the containment thermal response has been made.

4.1.1.2 Modelling Methodology

74 In accidents such as LOCAs, correlations and models used to represent physical conditions like the condensation need to capture all the appropriate forms of the various water/steam phases in the proximity of the walls, and numerous thermal capacity effects of large components including liner and concrete shells that will make an important contribution towards controlling pressure and temperature rises. These representations of heat structures within the codes have a major impact on conditions challenging the containment and the resultant natural convection phenomena in the containment environment is critical to controlling the containment's pressure and temperature demands during accident scenarios.

Assessment

75 Steam condensation is a complex phenomenon that is considered important with respect to the passive nature of heat transfer associated with the complex structures of the containment. Containment flow distributions within the large volume and sub-compartments are not captured within the modelling to a high accuracy level using lumped parameter type codes. As a result the heat transfer coefficients are likely to be variable on all structures. In particular, the steel liner will be subject to differing degrees of condensation depending on the prevailing local flow and temperature. In addition, there may be variable liner-to-the-concrete-shell conduction heat transfer effects also affecting the localised surface temperature distribution.

76 The computer codes employed by EDF and AREVA are:

- CATHARE – for the primary system thermal hydraulics and energy transfer in the containment whose conditions are computed by CONPATE4.
- CONPATE4 which uses the CATHARE input conditions into containment and computes the prevailing conditions and passes the IRWST water temperature and containment pressure as boundary conditions to CATHARE.

77 The adequacy of CATHARE is examined in the design-basis fault area and discussed in the Fault Studies report, Ref. 32. Its application to determining the break flow into containment is within the scope of its general validation and is not considered further here.

78 The CONPATE4 code is a lumped-parameter representation of the containment and its use appears generally consistent with previous practice. This analysis has received significant attention by other regulators and my experience suggests that this type of model is generally conservative. I have therefore not chosen to examine it in detail.

79 Calculations have been performed using the CATHARE code to predict transient primary circuit conditions and energy releases into the containment (for LOCA fault scenarios and loss of offsite power) assuming a pre-defined containment pressure, followed by containment analysis using the containment code CONPATE V4.

80 The containment model provides the global containment pressure and Medium Head (MH) and Low Head (LH) Safety Injection (MHSI / LHSI) temperature – boundary conditions that are fed back to the CATHARE code.

81 EDF and AREVA claim that conservative assumptions are employed as the requirement is to determine the maximum pressure and temperature within the containment environment, and the maximum IRWST water temperature, which I concur. These assumptions are maximum decay heat, maximum core power generation of 102%, loss of one diesel and no credit for manual action.

82 The details of the code coupling have been assessed in the Fault Studies discipline assessment report, Ref. 32.

83 In response to a query from the Fault Studies Assessors, EDF and AREVA have indicated an intention to provide UK specific studies during the detailed design and nuclear site licensing phases. I welcome this and note the related Assessment Finding, **AF-UKEPR-FS-20** (see Ref. 32), seeking an assurance that this commitment is completed at a later phase.

4.1.2 In-Containment – Unintended / Undetected opening of Foils and Dampers

Assessment

84 The EDF and AREVA access philosophy within the containment considers the operational requirements of the plant during normal operation, shut down conditions, planned outages and accident conditions. Access requirements are part of the general ventilation management system, which provide an overview of the ventilation management in normal operating and accident conditions. The ventilation strategy of the containment has not been included in the sampling of the PCSR at Step 4. It should be noted that sample assessment of parts of the overall ventilation system has been performed by the Mechanical Engineering discipline and it is reported in Ref. 29.

85 The two separate regions within the containment are created by the presence of passively operated flaps and dampers that are strategically located in order to allow access for maintenance in the accessible areas. During abnormal or fault conditions (increased pressure or temperature within the containment) these foils and dampers open to create a single large volume to allow improved mixing of the potential steam or gases released into the containment atmosphere, and increased surfaces for heat transfer and condensation.

86 I have not examined the reliability of the foils and dampers during normal operating conditions as part of my GDA Step 4 assessment as there is a lack of operational data and visibility of supporting test results. The failure of these components in normal operating conditions could reduce the effectiveness of the two room concept. In addition, the undetected failure of these components could create the possibility of convecting contamination into the accessible areas exposing any operators to airborne radionuclides. The adequacy of the in-containment radiation monitoring system falls within the assessment remit of the C&I discipline of the ND, and it is reported in Ref. 33 that the in-containment monitoring system is out of scope for GDA and will be covered during the site specific phase.

Findings

87 There are a number of observations made with regards to the overall ventilation philosophy during normal operating and fault conditions relating to the foils and dampers which are responsible for the separation of the two atmospheres within UK EPR containment. I therefore consider that the licensee should update the generic PCSR to provide a robust justification for:

> ***AF-UKEPR-CSA-01*** *- The licensee shall, prior to inactive commissioning – containment pressure test, provide the ventilation strategy supporting the concept of inaccessible/accessible areas during normal operations and accident conditions for situations where one or more of the foils and dampers have failed.*
>
> ***AF-UKEPR-CSA-02*** *- The licensee shall, prior to inactive commissioning – containment pressure test, provide the test results to support the claims for the performance and the reliability of the foils and dampers used in the CONVECT system.*
>
> ***AF-UKEPR-CSA-03*** *- The licensee shall, prior to inactive commissioning – containment pressure test, provide clarification of the impact of the availability of the foils and dampers on plant operation and specifically, how this is controlled by technical specification.*
>
> ***AF-UKEPR-CSA-04*** *- The licensee shall, prior to inactive commissioning – containment pressure test, provide analysis to examine the impact of unintended and/or undetected opening of the foils and dampers on the pressure and temperature monitoring informing the accident management procedures.*
>
> ***AF-UKEPR-CSA-05*** *- The licensee shall, prior to inactive commissioning – containment pressure test, provide analysis to examine the impact of incomplete operation of the CONVECT system.*

4.1.3 Containment Isolation

Assessment

88 The internal containment with its leak-tight steel liner together with its pre-stressed concrete walls provides the pressure boundary for the containment. The steel liner ensures the functionality of leak-tightness whilst the resistance to increased pressure demands is met by the pre-stressed concrete walls. The containment design limits are given in the PCSR as 5.5 bar and 170^{o}C. The various DBA scenarios are analysed to check for compliance that these design limits are not exceeded. The design includes additional margin to failure due to the inherent margins in the design method. In addition,

EDF and AREVA have generated fragility values for use in the PSA studies, which has been assessed by colleagues from the PSA discipline and reported in Ref. 34.

89 The UK EPR containment during accident scenarios experiences various thermal loadings and it has to manage the containment isolation system together with the combustible gas control system. The demands are caused by the environmental and dynamic effects associated with all aspects of plant operation, including those arising from design basis accidents and beyond design basis accidents.

90 This pressure boundary is penetrated by a number of pipes required for normal operation of the plant and all non-essential penetrations include isolation valves which prevent the containment boundary being bypassed in the event of an accident.

91 The containment isolation valves are either locked closed or are closed automatically in the event of an accident except where systems are required for accident management such as Safety Injection Systems (SIS). Various systems are dedicated to severe accidents and will not be employed during DBA.

92 The various fluid/gas penetrations include; main steam-lines, feed-water systems, numerous pipe penetrations, sump suction penetrations, and ventilation system penetrations. Although information relating to the design of these penetrations and isolation systems was available during the GDA, I did not assess this in any detail. However, I note that the functional intent of the design appears to be in accordance with relevant good practice elsewhere.

93 However, the performance of these isolated fluid/gas systems and the related penetrations through the containment should be further examined when more detailed design information becomes available during site licensing. This leads to the Assessment Finding below.

Finding

AF-UKEPR-CSA-06 *- The licensee shall, prior to active commissioning – cold operations, justify that the isolation systems and containment penetrations meet the site specific loading requirements (pressure, temperature, moisture and leakage) in accident conditions.*

4.1.4 Containment Sump Performance

94 In design-basis faults, reactor coolant inventory is generally replenished by safety injection from the in-containment refuelling water storage tank. However, this vessel has a limited size and ultimately will empty. In the largest loss-of-coolant accidents, this can happen in a matter of hours. Under these circumstances, the operator is required to realign the injection pump suction lines to take water from the containment sump.

95 It is necessary to ensure that, should this occur, debris in the containment building is not swept into the primary circuit where it would impair cooling. Consequentially EDF and AREVA have designed a complex system of sump strainers. These are intended to limit the ingress of debris into the primary circuit, while allowing sufficient flow through to ensure adequate functioning of the safety injection pumps.

4.1.4.1 EDF and AREVA's Case

96 In the event of a pipe failure of the Reactor Coolant System, water can carry debris to containment sump strainers. The main debris is derived from insulation material used to cover the RCS. Other debris to be considered after a pipe failure consists of paint chips, latent debris, and concrete dust.

97 The insulation (fibre glass) covers main parts of the RCS (Main Coolant Lines, RPV-head, Steam Generators, Pressuriser) and the auxiliary piping. To limit its detachment and transportation in the flow, mineral wool has been encapsulated in reinforced cassettes which are mechanically attached on the RCS components.

98 Cassettes are generally mounted in a mattress with reinforced metallic protection or by metallic sheet.

99 Microporous material (i.e. Microtherm) is only used where space is limited. To limit the jet-impact effect, e.g, from a fluid discharge due to a Loss of Coolant Accident (LOCA), microporous material has been encapsulated in reinforced cassettes.

100 The above measures have been taken into account to limit the jet-impact effect and thus the amount of insulating material detached and transported in the flow. However, the injection system is equipped with a complex passive filtration system designed to keep the injection system operational even with a large amount of debris. A conservative approach has been used to define the assumptions for its sizing.

101 The passive filtration system is constituted of the following parts:

- Wall-weirs and trash-racks located at “heavy-floor” openings. Their function is to allow deposition of debris out of IRWST (sedimentation).
- Retention baskets located below the heavy-floor openings. Their function is to keep large debris out of IRWST (by means of robust bars and large mesh size).
- Further set of retention baskets located below heavy-floor openings. Their function is to retain 50% of debris in case of LOCA (by mean of a large retention volume).
- Strainers located above IRWST sump pit on the SIS pump suction and CHRS pump suction.

102 Consequently, only a small amount of debris (in particular divided fibres coming from insulating material) goes through the sump strainers - limiting the risk of the clogging of safety injection equipment or fuel assemblies.

Assessment

103 In TQ-EPR-1053, I requested that EDF and AREVA outline the measures taken to ensure that insulating material, used to cover the Reactor Coolant System, can not become detached and transported in the flow as finely divided fibres in the event of a pipe failure of the primary coolant system. The response to this request has come late in GDA and therefore I consulted with other regulators. I was advised that EDF and AREVA have carried out a substantial amount of development and tests on this issue and it is expected to be resolved satisfactorily, but the assessment is continuing.

104 The response I received detailed measures taken to qualify the design, but did not detail the options reviewed before reaching the design offered and did not demonstrate why it is ALARP. I note: Firstly that the issue of mineral insulation is avoided at Sizewell ‘B’ by the extensive use of metal foil insulation. Secondly, it appears that the insulation offered

differs in different countries. EDF and AREVA have not provided a satisfactory explanation for this.

105 The assessment of fibrous debris on safety injection equipment and its qualification in accident conditions has been performed in the Mechanical Engineering topic area (Ref. 29) with an extant Assessment Finding requesting that the qualification tests are satisfactorily completed to demonstrate the adequacy of the performance for the equipment in the expected operational conditions.

106 In the case of the fuel, the design of the fuel assemblies is such that most debris is likely to be removed at the inlet nozzles, and in any case the flow quickly redistributes in the event of local blockage of the inlet of a limited number of fuel assemblies. Significant amounts of fibrous debris entering the fuel would prevent its reuse and would make the case for restarting the reactor difficult to accept. However, a substantially complete blockage would be required to prevent fuel cooling post LOCA.

107 Overall, I expect that the UK EPR project should identify a design which reduces risks in this area as far as reasonably practicable and I feel that a common position between regulators internationally is desirable. I am therefore raising an assessment finding requiring that a potential licensee demonstrate why the proposed design is ALARP.

Finding

> ***AF-UKEPR-CSA-07*** *- The licensee shall, prior to inactive commissioning – containment pressure test, demonstrate that the design of insulation and the strainer structures associated with the safety injection system is such that the risk of sump blockage has been reduced to the lowest level reasonably practicable. In particular, the licensee should produce an analysis of the options and justify the choice of insulating technology*.

4.2 Effectiveness of the Measures to Depressurise Reactor Coolant System

4.2.1 Core Outlet Temperatures

108 The operator may depressurise the RCS at various stages during the fault conditions, but not whilst at power. However, depressurisation is anticipated to be activated by the operator when the Core Outlet Temperature (COT) reaches 650°C. The core outlet temperature is also proposed to be used for initiation of severe accident management procedures associated with control of debris and containment performance. The objective of this is to ensure that the primary system is depressurised prior to relocation of molten corium to the vessel lower head and a consequential failure of the vessel.

Assessment

109 The effectiveness of the measurement of COT in accident management was reviewed by the CSNI working Group which concluded (Ref. 35) that a combination of a selection of core outlet temperature readings and other instrumentation indications, such as reactor vessel water level, should be used to define the initiation of the different accident management procedures. Ref. 35 indicates that various test results suggest the thermocouple responses significantly lagged behind the cladding temperatures. This brings into question the effectiveness of this measure as a way of preventing core melt, but since this is not the objective in the UK EPR strategy, I consider that this delay does

not significantly impact the time available to act and prevent a high-pressure vessel failure.

110 It may be ALARP to initiate depressurisation earlier using alternative indications such as core water levels as a means of preventing core damage earlier in the event.

111 I recognise that the core outlet temperature measurement is supported by redundancy and diversity of other instrumentation effectively measuring the COTs via for example hot leg thermocouples. However, I consider that in the light of the experimental data provided by Ref. 35, it is necessary to raise an Assessment Finding regarding the accuracy of the measured coolant temperatures in such conditions.

112 I note that the thermocouples measuring core outlet temperatures are routed via the RPV head. The routing of such instrumentation that is used to inform accident management procedure is potentially at risk from fault scenarios, such as excessive corrosion around the nozzles housing the Control Rod Drive Mechanism (CRDM). The loss of coolant from such locations could lead to a direct steam impingement onto these instrumentation lines during accident conditions, and is likely to adversely impact on the availability/reliability of the instrumentation that are routed/supported by the RPV head. The corrosion experience at Davis-Besse plant reinforces the importance of protecting such instrumentation lines routed via the RPV head.

113 I do however recognise that the occurrence of fault conditions needing such instrumentation is a low probability event, and the lessons learnt following the Davis-Besse plant incurring CRDM corrosion will be considered within the maintenance requirements of the UK EPR plant.

114 In summary; in fault conditions where the operator action is highly dependant on measurements such as COT output, other instrumentation such as the hot leg temperature measurement and other reactor temperature and water level indicators are available to the operator and should be considered through an holistic approach. I have raised the concern relating to the availability of such instrumentation informing any pending operator action with EDF and AREVA.

Findings

115 There are a number of observations made with regards to the operational requirements for instrumentation indicating the on-set of a severe accident; given the significance of the instrumentation shortfall identified in Ref. 35;

> ***AF-UKEPR-CSA-08*** *- The licensee shall, prior to active commissioning – cold operations, justify the measurement systems indicating core conditions used to initiate the accident management procedures, such as, core outlet temperature measurements and the reliability of instrumentation routed via the RPV head; the justification should give consideration to common cause failure.*
>
> ***AF-UKEPR-CSA-09*** *- The licensee shall, prior to active commissioning – cold operations, provide an analysis of the impact on safety from degradation through ageing of the in-vessel thermocouples with a view of establishing maintenance plans assuring the integrity of this equipment over long operational periods and throughout the plant's lifetime.*

4.2.2 Primary Depressurisation System (PDS) Prior to Severe Accidents

116 In addition to other functional requirements claimed in design-basis faults such as Feed and Bleed (F&B), RCS depressurisation is achieved through a dedicated route – Primary Depressurisation System - from the pressuriser that is independent of the pressuriser safety relief valves. The coolant is discharged into the Pressuriser Relief Tank (PRT), which itself is protected by rupture disks which discharge into the containment.

117 The valves are intended to discharge a mixture of water and steam at high flow rates to rapidly depressurise the RCS to below 20 bar which relates to the cases of late re-flood. It is however, recognised that based on best estimate predictions a lower pressure (below 5 bar) may be achieved for other cases.

118 In the context of severe accidents, the primary depressurisation system aims to avoid the possibility of High Pressure Melt Ejection (HPME) and the potential for Direct Containment Heating (DCH), phenomena which can lead to early containment failure. Chapter 16 of the PCSR states that the containment design takes into account consequences related to a severe accident but without considering loads induced by a high pressure melt ejection, i.e, rupture of the reactor coolant system at high pressure is excluded by design.

Assessment

119 The successful activation of the PDS on demand in the accident scenario is a key step in accident management. The operator may depressurise at various stages during a fault but depressurisation will eventually be activated by the operator when the core outlet temperature reaches 650°C. The manual operation introduces a degree of uncertainty into the time and rate of depressurisation and affects the performance of the accumulator supplying water to fuel at high temperature causing rapid zirconium oxidation, which potentially results in high rates of hydrogen generation. The impact of this on containment integrity has been assessed by EDF and AREVA and found to be satisfactory. In a recent response (TQ-EPR-1388, Ref. 9), EDF and AREVA have indicated that a full risk assessment has been performed to examine the PDS operation and concluded that manual operation of the PDS is preferred to that of automatic initiation.

120 A key to this outcome is the risk of inadvertent operation of the system at pressures likely to degrade the plant safety, and EDF and AREVA have stated that the reliability level required to support an automatic actuation of PDS system is very hard to justify in the current PSA model. For this reason the decision was taken to have a manual actuation of the valve.

121 I have assessed the arguments and the risks associated with inadvertent operation of the PDS valves at high pressure and dual usage for feed and bleed to reduce the RCS pressure prior to a severe accident developing. EDF and AREVA believe it is not ALARP to automate, and I have accepted this argument. However, given the complexity of the arguments and the potential safety dis-benefit, I consider that the mode of PDS operation and the role of the operator should be reviewed within the Emergency Operation Principles (EOPs) and the OSSA as part of the licensing activities.

122 In summary, the generic PCSR does not fully describe the functional requirements of the PDS during design basis and severe accidents. The successful initiation of the PDS is a key step within the severe accident management procedures in preventing high pressure accident scenarios leading to a HPME. I also note that there are complex interactions between the OSSA expectations, human factors, the accident transient and the

implementation of the system. I have therefore raised the following Assessment finding requesting a full justification of the operational strategy of this system.

Finding

> ***AF-UKEPR-CSA-10*** *- The licensee shall, prior to active commissioning – cold operations, provide a robust justification of the operational requirements of the PDS during fault conditions. The justification is expected to fully consider the PDS implementation and Operating Strategies for Severe Accident (OSSA) for the UK EPR.*

4.3 Severe Accidents with Failure to Restore Cooling

123 The severe accident generally evolves from a loss of core cooling capability leading to fuel degradation, and core melt that may eventually lead to fuel relocation. The measures in place to mitigate the consequences are limited in most existing plants, but HSE's safety assessment principles require that reasonably practical measures are taken to limit the consequences of such events.

124 In most new reactor designs, the widely adopted objective of "practically eliminating large early releases of radiation from the containment", has led to engineered features dedicated to retention of molten core debris. However the strategy has been the subject of debate for some time, with no single optimum measure becoming apparent.

125 The UK EPR adopts the option of providing a spreading compartment to retain the core debris in a suitable configuration for stable long term cooling. The main features of the CMSS intended to facilitate this are illustrated in Figure 1. This assessment is unable to determine which of the strategies for managing a molten core is optimum, but instead considers whether a suitable and sufficient safety case has been presented.

126 Given the low likelihood of a severe accident, the safety assessment principles do not require a formal pessimistic design-basis assessment of the success of mitigation measures. The requirement is to perform best-estimate analysis to demonstrate the measures taken are likely to be sufficient to reduce the risk to acceptable levels.

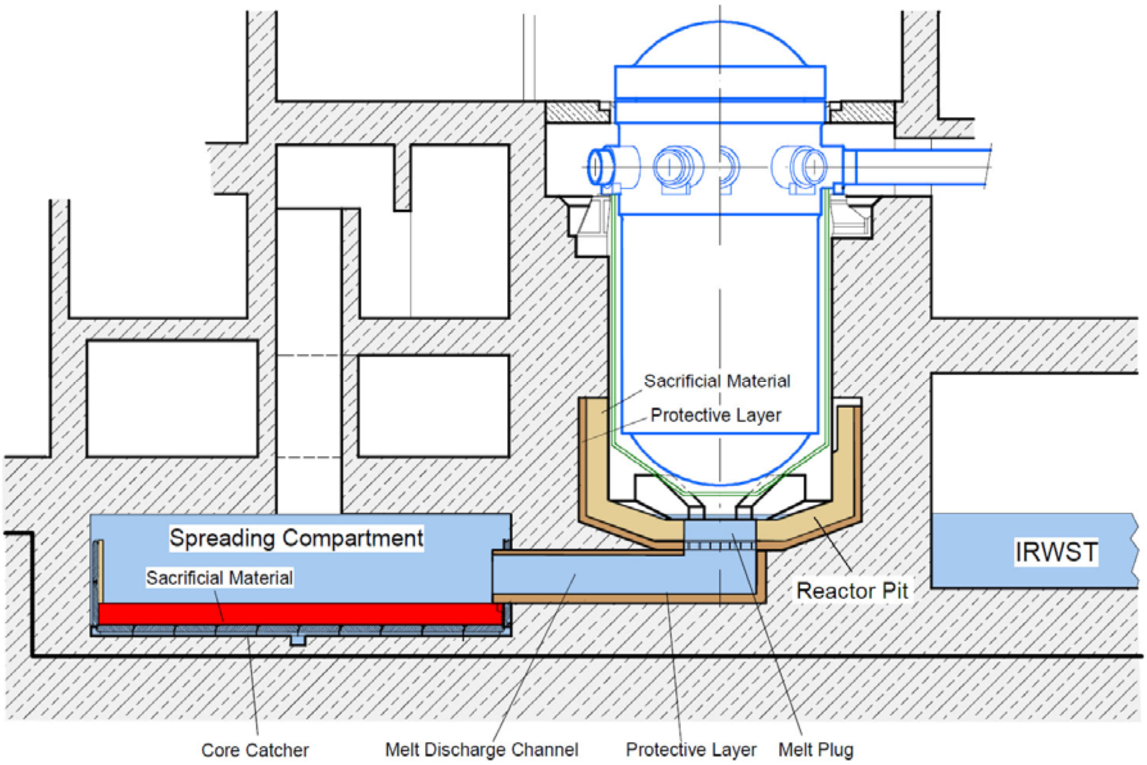

Figure 1: Schematic View of CMSS Main Features

4.3.1 EDF and AREVA's Case

127 Chapter 16 of the PCSR presents the analyses that characterise melt processes including an examination of the phenomena occurring that could challenge the plant. The justification is based on a combination of international specific research experiments and detailed computational modelling.

128 In cases where the onset of core damage can not be prevented despite the depressurisation of the primary circuit, EDF and AREVA argue that the benefits of attempting to inject cooling water to re-flood the already damaged core at any time are outweighed by the risks associated with late re-flooding with vessel failure. EDF and AREVA therefore plan to permit the core damage to progress to core melt and to catch the molten core in a purpose-designed facility where it can be held in a stable configuration and cooled.

129 Analysis indicates that in the absence of core cooling, molten core material will fall to the bottom of the reactor vessel where it will initially encounter some residual water and will temporarily re-solidify into a debris bed. There is a possibility of a steam explosion as a result of interaction between the molten fuel and the coolant, but the water is not likely to be sub-cooled and the process will be gradual, so it is argued that the interaction will be sufficiently low energy not to lead directly to vessel failure.

130 As debris accumulates at the bottom of the pressure vessel, it will eventually re-melt as the remaining water is evaporated. The resulting melt will begin to ablate the walls of the vessel lower plenum, leading to vessel failure and release of the molten material into the reactor vessel pit.

131 The composition of the corium collected within the lower head comprises mainly of oxide and metallic components with the oxide being the significant part. On the basis that there is likely to be some separation and stratification of the metallic and oxidic layer, EDF and AREVA argue that the failure is likely to occur in the side of the vessel and the melt is likely to pour into the vessel pit progressively as the vessel lower head collapses. If no such layering takes place, failure is also expected towards the top of the molten pool since natural convection will lead to higher heat flux densities to the RPV wall near the top of the pool.

132 After RPV failure, the molten corium is intended to first accumulate in the vessel pit and later transfer, in one rapid pour, into its final configuration in the spreading compartment.

133 A period of melt retention in the reactor pit addresses the view that the release of molten material from the RPV bottom head will, most likely, not take place in a single release, but over a period of time that may be less than two hours. The interaction of the debris with sacrificial concrete affects the melt composition, so that its temperature and viscosity are appropriate allowing the debris to flow through the transfer channel. An aluminium gate releases the melt.

134 Spreading of the melt into the core catcher located at the base of the spreading compartment will be followed by flooding, quenching with water, and sustained cooling of the core debris.

135 The intent of the design is that the molten material will be spread sufficiently evenly that it can be cooled efficiently and retained in a stable configuration where it can not damage the structure of the containment building. The design is also such as to minimise the release of gas from concrete materials as a result of melt-concrete interaction inside the core catcher.

4.3.2 Overview

136 The design involves novel provisions for the retention and long-term stabilisation of the molten core inside containment in the highly unlikely event of a severe accident. After RPV failure the molten corium is intended to first accumulate in the reactor cavity and later further relocate, in one event, into a lateral spreading compartment. Spreading of the melt will be followed by flooding, quenching, and sustained cooling of the corium.

137 Many debris management options have been evaluated by EDF and AREVA within the design evolution prior to the final design being offered for the UK EPR reactor. This design corresponds in its entirety to that being built at Olkiluoto 3 in Finland and Flamanville 3 in France. I therefore have been able to benefit from assessment carried out in these countries.

138 The assessment of core melt management is considered in a sequence of distinct phases as follows:

- Core degradation;
- RPV bottom head failure and melt release; and
- Melt stabilisation.

139 The assessment review is presented for each of these phases in the following sections.

4.3.3 Core Degradation

140 The severe accident generally evolves from a loss of core cooling capability and fuel overheating leading to core melt and potential relocation. There are complex phenomena and physical relocation as to how the degradation phase, failure stage and relocation of the core into the reactor pressure vessel bottom lower head occurs. The supporting arguments presented are largely based on computer code work using the Modular Accident Analysis Programme (MAAP) code. The manner of the loss of core coolable geometry and debris relocation is determined by the MAAP code to be either sideways and downwards or only downwards through the lower support plate. It is a function of the melting temperatures of all the core components and the non-solidified pathways for the debris transport that are available.

141 The debris that does relocate into the bottom head is subject to complex thermal hydraulic phenomena, melt composition and chemistry, and melt coolant interactions which make its behaviour subject to degrees of uncertainty. The type of failure of the bottom head impacts on the flow of debris into the next stage of corium debris management which is the reactor pit. The bottom head structural integrity response in terms of wall ablation, creep rupture dictates how the debris will outflow from the head.

4.3.4 RPV Bottom Head Failure and Melt Release from RPV

142 The RPV bottom head failure results from two phenomena of importance; the composition of the core debris relocated and the separation of metallic and “oxidic layers”. The type of failure can be a laterally located opening, similar to a ‘fish-mouth’ as observed in the FOREVER tests, or some other type such as a hole at the bottom of the head.

143 The molten debris remains within the lower head for a significant period of time and various heat transfer processes in particular radiation will eventually melt all the material inside the vessel. The location of the vessel failure dictates the pours to the reactor pit. If it fails with a bottom central hole there may be a single long duration pour. If the lower head fails laterally, there may be a pour from this location and then a later pour from the final failure of the RPV to release all the liquid debris into the reactor pit.

144 Debris behaviour for PWR plants is usually based on both research and TMI2 accident insights which showed separation of lighter metallic and heavier ceramic metal layers, and no failure of the RPV bottom head. No evidence of specific EPR material composition test insights in the hot pool Russian RASPLAV experiment is found in documentation or from any other facility. Thus, the behaviour of the debris in the head will be based on generic considerations of composition and thermal hydraulics insights supported by computer code modelling.

145 Even if the modes of RPV bottom head failure were to be different to that postulated from the experimental tests, such differences in accident progression are claimed by EDF and AREVA to be removed by the residence time in the reactor pit. This allows time for all remaining and applicable solid material to become molten within the reactor pit volume. This residence time will ameliorate many of the unknowns and uncertainties that may prevail with respect to the plant’s behaviour during the timeframe of RPV bottom head failure.

146 It should however be noted that the overall assessment of the structural integrity of the RPV is reported in Ref. 36, and the chemical composition of the melt contained in the RPV lower head is covered by the reactor chemistry discipline and is reported in Ref. 26.

147 In summary, international research is continuing to further improve the understanding of the core melt behaviour in fault conditions. The research is linked to international initiatives to improve the code predictive capabilities in an effort to reduce the uncertainties associated with modelling, and capturing the complex phenomena associated with these conditions. I acknowledge that EDF and AREVA have been active in performing research and development in support of areas relevant to the generic EPR. I would therefore, strongly encourage the future licensee to continue maintaining their involvement and support these initiatives to be an informed licensee.

4.3.4.1 RPV Failure Location and Melt Dispersal

148 EDF and AREVA have provided information regarding the RPV failure location and its mode of failure (Ref. 12 Sub-chapter 16.2). The failure can be at the centre of the lower head or laterally part way up the head. In the former, it could result in a “single pour” mode through a central hole in the bottom head. In the latter, the failure produces an initial pour with partial release which reduces the volume of melt to the failure height, and then in the worst case scenario partial or complete head failure may occur. The “two-pour” scenario is seen as quasi continuous molten corium pouring between the first and second pours.

149 EDF and AREVA have also provided clarification (Ref. 16) in order to justify that the initial RPV lower head failure location would likely be laterally with a higher probability than failure at the central lowest location. This appears to be consistent with the results from the relevant experimental data.

150 In addition, EDF and AREVA claim credit from DISCO experiments with EPR type geometry which considered several test cases of RPV failure to determine the mode of debris release from the RPV lower head. The high discharge capacity of the PDS lines is expected to reduce the RCS pressure to below 5 bar at vessel failure time for the most likely scenarios. In such cases, independent of the uncertainties relating to the location of the RPV lower head failure, the probability of significant melt dispersal is expected to be negligible. However, in case of a scenario with a late re-flood, at the time of the RPV failure, the pressure could be up to 20 bar. The tests showed that the amount of corium dispersed out of the reactor pit when central bottom failure occurs was in the order of ~40%. This quantity was a combination of ~35% (liquid discharge) transported to the surrounding compartments and ~5% (gases and small particles) dispersed into the containment atmosphere, leading to direct containment heating. When the failure is located laterally on the side wall of the RPV bottom head, the dispersed fraction is much lower.

151 EDF and AREVA have considered a gaseous phase dispersal fraction into the containment atmosphere that is higher than that observed from the DISCO experiments. I accept that this represents a conservative assumption.

152 EDF and AREVA have however not considered the potential liquid phase fraction which could be dispersed into the surrounding compartments via paths adjacent to the RPV nozzles in the PCSR. This is because the RPV bottom lower head failure is predicted to occur laterally using the MAAP code. There are significant uncertainties remaining in predicting failure location of the RPV as a result of in-vessel complex thermal hydraulic, chemical and structural interactions related to the relocated corium. Considering these uncertainties, should a less likely central bottom head failure occur, a significant amount of the core melt could be transported out of the reactor pit into the adjacent compartments. This would not be treated by any dedicated cooling system and might

interact with the structural concrete. This would also be likely to affect the containment atmosphere.

153 In order to develop confidence in the design of the reactor pit to accommodate the molten material in adverse pressure conditions (late re-flood), I raised a query (TQ-EPR-1413, Ref. 9) relating to the configuration of the flow path between the reactor pit and the adjacent compartment. In its response, EDF and AREVA indicated that the potential flow path between the reactor pit and the adjacent compartments close to the RPV nozzles is relatively small, effectively limiting the total flow. The response does not however provide a sufficient demonstration that conservative arrangements have been adopted to justify the claims for this limiting effect.

154 In light of the results of the DISCO experiments, I have raised the following Assessment Finding requesting validated evidence to demonstrate that reasonable measures are adopted to retain the molten material within the reactor pit in case of central bottom RPV failure relating to late re-flood conditions.

Finding

> ***AF-UKEPR-CSA-11*** *- The licensee shall, prior to construction – nuclear island safety-related concrete, provide validated evidence that either potential release of molten material from the reactor pit into the adjacent compartments is as low as reasonably practicable for the cases of central bottom RPV failure relating to late re-flood conditions, or that the melt release does not lead to the loss of containment integrity*.

4.3.4.2 Height of the Dedicated Features in the Reactor Pit

155 The reactor pit design is intended to accommodate and retain the volume of molten material that may be available for relocation from the RPV. The reactor pit is illustrated in Figure 2. The lower surfaces within the reactor pit are covered by sacrificial concrete laid above protective ceramic layers. I have considered the total amount of corium that could accumulate within the RPV lower head and how this relates to the height of the protective ceramic layer in the reactor pit. The concern here related to the potential failure level of the RPV that may be higher than the height of the sacrificial concrete and protective layer.

156 In response to an enquiry (TQ-EPR-1060, Ref. 9) EDF and AREVA have provided information relating to the likely height that the RPV will fail at due to the two competing phenomena; relocation of corium within the RPV, and RPV wall creep failure. The combination of thermal, chemical composition and structural effects is claimed to influence the creep failure, and the timing of the failure will influence the quantity of the corium being relocated into lower head and hence the associated location of the RPV wall failure.

157 I recognise that the RPV failure is an energetic event, combining significant quantities of core melt at high temperature, and the failure phenomena and supporting analysis to predict the melt behaviour are complex. I thus consider the timing and precise RPV failure dynamics contain uncertainties that are dependent on accident scenarios, assumptions and simplifications associated with the modelling capturing the key phenomena. I accept the response to TQ-EPR-1060.

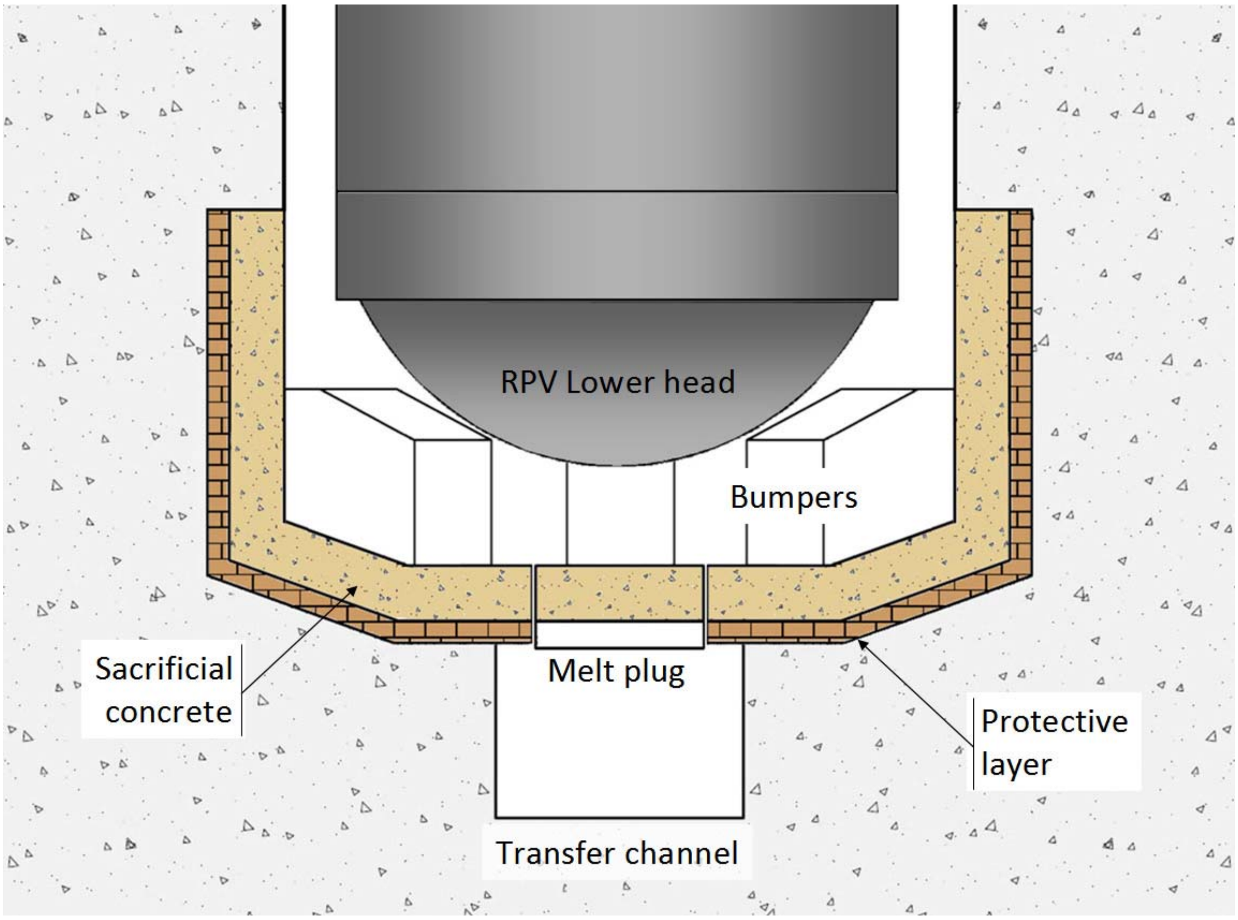

Figure 2: Outline View of the Reactor Pit

4.3.5 Core Melt Stabilisation System (CMSS)

158 The processes in the CMSS are discussed individually by the following items:

- background and rationale to CMSS development;
- melt collection and molten core concrete interaction within the reactor pit;
- melt plug operations (failure / opening);
- flow of corium into the spreading compartment;
- IRWST passive injection into the spreading compartment;
- initial corium cooling due to flooding; and
- long term cooling of the melt in the spreading compartment.

4.3.5.1 Background and Rationale to CMSS Development

159 EDF and AREVA, in a technical meeting, have presented the overall development approach for molten debris confinement, and provided their justification for the key

fundamental features of the CMSS, although this is not explicitly reflected within the PCSR. In the process, concepts such as the strategy of In-Vessel Retention (IVR) and an Ex-Vessel core catcher located below the RPV have also been examined. EDF and AREVA outlined the various reasons for the development and selection of the proposed CMSS design having considered technical challenges, such as Critical Heat Flux (CHF) for external cooling of the RPV wall for the predicted decay heat levels, and the possibilities to manage the potential risk of steam explosion.

160 CMSS is a design feature with an objective to facilitate the controlled movement of debris away from the reactor to another location. Important aspects of CMSS are that it provides for conditioning of the melt and helps compensate for uncertainties. Thus, in the event of a severe accident and core melt occurring, the melt has the potential to relocate in the RPV bottom head until it fails when it will relocate to the next stage and flow to the reactor pit. A period of melt retention in the reactor pit will occur and the need for this temporary retention addresses the view that the release of molten material from the RPV bottom head will, most likely, not take place in a single release, but over a period of time that may be a couple of hours. The interaction of the debris with the concrete within the reactor pit is to 'condition' the melt to allow it to flow down into the transfer channel.

161 Accumulation and temporary retention within the reactor pit is claimed by EDF and AREVA to be assured through the layer of sacrificial material that must be penetrated to escape into the transfer channel. This delay ensures that, in case of an incomplete first release of melt from the RPV, practically the entire core inventory will be collected in the reactor cavity and be fully molten at the time of gate failure. Heat transfer processes from the melt cause the un-molten material in the RPV lower head to melt.

162 The combined mixture of the sacrificial material and molten debris dictates the spectrum of possible melt states prior to further spreading and makes the melt properties (and, therefore, subsequent stabilisation measures) independent of the uncertainties related to the initial release of melt from the RPV.

163 The retention time in the reactor pit is primarily driven by the thickness of the concrete cover and not by the delay-to-failure time of the gate after melt contact. The gate is the lowest point within the reactor pit and the only location where the sacrificial concrete is not backed by a protective layer. Therefore, the melt plug and retention gate represents the pre-defined failure location for melt retention in the cavity. Following the failure of the gate, the melt will progress through the transfer channel in a continuous pour and enter the spreading compartment.

164 Due to its large cross-section and its temperature-resistant walls, the transfer channel itself is expected to have no retarding effect on the flow. Under the predicted outflow conditions, blockages at the melt front in the transfer channel cannot occur, even for purely oxide melts. While metallic melts are reported to spread '*like water*' as they generally have a very low viscosity and a high heat capacity, an initial liquid oxide melt can only solidify during contact with the walls of the channel. As these consist of zirconium dioxide (zirconia), which has a low thermal conductivity, the amount of heat that can be absorbed is very low. Given the propensity for the oxides to contain fission products, internal decay heat generation provides an inherent limitation in the cool-ability of the melt through the transfer channel. In addition, debris that might flow with the melt has a limited affinity for attachment to the walls. All the above are good arguments for assuring the flow of corium after the melt plug/gate's defined failure.

165 During the initial process of the melt entering the spreading area, spring loaded water flooding valves will be opened by a thermal actuator. Through these valves, water flow,

driven by the head of water within the IRWST, passively flood an array of horizontal cooling channels formed by the '*fins*' extending from cast iron cooling elements below a layer of sacrificial concrete that ultimately supports the melt spread. With the initial flooding rate of 100 kg/sec the filling process will be completed in approximately 5 minutes. The flow of water onto the melt will then continue from the circumference of the spreading room. This water flow will continue until the hydrostatic pressure levels of the IRWST and spreading room are balanced. Molten metal is claimed to solidify in a few hours and molten oxide material in a few days due to internal heat generation from radioactive decay.

166 Water overflow into the spreading room covering the melt will continue until the column of water is balanced by the column of the water within the IRWST. This results in the submersion of the spreading area and transfer channel as well as a portion of the reactor cavity. This stabilises all residual core debris in all areas.

167 The continuing stabilisation of the melt is based on cooling and crust formation. Due to the high surface-area-to-volume ratio created by the spreading process and the fact that the melt is completely surrounded by cooled surfaces, an enclosure of the molten core debris within a crust envelope is expected to be achieved soon after the end of the molten core-concrete interaction in the spreading area.

Assessment

168 The complex physical and chemical processes involved during the transient covering the phases from the RPV failure to the long-term cooling of the corium in the spreading compartment has been examined separately. I have considered each phase with its specific safety objectives and dedicated design arrangements aimed at assuring its success. I have been assured it will manage the debris relocation in a controlled manner by minimising the risk of steam explosion and by conditioning the melt to have appropriate physical properties to flow un-heeded through the transfer channel between the reactor pit and spreading compartment.

169 I have considered the arguments in support of the ALARP approach presented by EDF and AREVA. I am aware that much of the justification is dependent on the interpretation of experimental separate effects tests for different aspects of the CMSS, and the use of deterministic analyses using computer codes. EDF and AREVA have stated that the COSACO code predictions which includes the CMSS aspects covering reactor pit and core spreading area is supported by an appropriate code validation, and should be preferred for its insights ahead of MAAP predictions.

170 Assuming that any revised calculation with an updated model in COSACO (to capture anisotropic ablation) would not impact on the current tendency; I am satisfied with the concept of melt collection to perform its required duties. I acknowledge my judgement is influenced by insights obtained from experiments representing very complex phenomena. The validation of COSACO is further discussed in the codes and methodology, Section 4.7.

171 Considering that this area is currently the subject of ongoing international research, I expect more information will be provided on aspects of the various experimental facilities and their insights during the site licensing activities. Similarly, the relationship between experimental and plant conditions, overall linkage of codes for the different aspects of CMSS and the related supporting code validation and accuracy levels achieved will also be provided.

4.3.5.2 Melt Collection and Molten Core Concrete Interaction within the Reactor Pit

172 The MCCI takes place where the released corium comes into contact with the concrete present within the reactor pit. This is a design feature to allow for the collection and conditioning of the corium prior to the melt plug failure. The objectives of this aspect of the CMSS process are:

- The reactor pit shall retain the corium for a sufficiently long time such that all the melt transferred from the core and other structures are fully molten and collected before the melt plug opening.
- In addition, this will also help achieving a complete oxidation of the zirconium (Zr) inventory, a sufficient amount of oxidizing the totality of components dissolved within the core melt shall widen the solidus-liquidus temperature range in order to facilitate it having the appropriate characteristics such as viscosity and density. Regarding the density, the molten core concrete interaction shall provoke a layer inversion between the oxide and the metallic layers.

Assessment

Structural Stability

173 The reactor pit structure has to be able to withstand the resultant loads arising from an energetic release of significant quantities of core melt at high temperature.

174 In response to a query, (TQ-EPR-1069, Ref. 9), EDF and AREVA have provided information to justify the effectiveness of structural stability of the reactor pit in the presence of relocated corium. The response is also supported by an analysis that has considered a postulated degradation of the structural concrete (beneath the protective layer) of 10 cm. I have reviewed the results of this analysis and I am therefore content with the response.

Reactor Pit Bumpers - Function

175 The bumpers located in the bottom of the reactor pit are to protect the melt plug from the collapse of the RPV bottom head, and provide sacrificial concrete for conditioning of the melt. The bumpers will also reduce the free space within the reactor pit limiting the potential for steam explosion.

176 A key function of the reactor pit bumpers is to assist in absorbing the shock loading resulting from a potential sudden collapse of the RPV bottom head containing the debris due to total “unzipping” mechanism (circumferential failure) and the thrust resulting from the dynamic loading. It is considered that this will be the most onerous demand in terms of pressure and dead weight from tonnes of molten debris. This assumes a successful primary system depressurisation has been completed prior to the RPV failure.

177 In response to queries, (TQ-EPR-1055 & 1329, Ref. 9), EDF and AREVA have provided the assumptions and results of the calculations to demonstrate the functionality of the bumper design. I consider the response to be adequate at this stage.

Oxidation Process within the Corium Pool

178 Metallic elements contained in the initial corium pool are mainly uranium and zirconium (Zr), from the fuel and the clad; but there are also others elements, from the structural internal parts of the RPV and the RPV itself. The metallic constituents are chemically active prior to oxidisation.

179 In response to an enquiry (TQ-EPR-1054, Ref. 9), EDF and AREVA provided detailed information regarding the oxidation processes of the metallic elements contained in the reactor pit pool. There are uncertainties regarding the initial quantity of non-oxidized metallic constituents within the core melt. Hence, the PCSR has made a number of assumptions of different scenarios, and has justified that the sacrificial concrete used in the reactor pit will supply an adequate quantity of material to enable complete oxidation of zirconium in the core debris.

180 EDF and AREVA have provided an evaluation of the energy produced by the exothermic oxidation reactions based on the numerical code used to model the interaction between core melt and concrete. They also infer that the dedicated sacrificial concrete composition meets other requirements, such as decomposition enthalpy and mechanical stability.

181 Although the assessment of the chemical aspects generally falls within the assessment remit of other disciplines within HSE ND, given the current information, I consider that the arguments relating to oxidation phenomena have been appropriately addressed.

Anisotropic Ablation within the Reactor Pit

182 The ablation process of melt on concrete is a key phenomenon for the retention and conditioning phase within the reactor pit. OECD MCCI and VULCANO experiments have shown that the MCCI ablation progress may not be isotropic. Depending on the composition of concrete, the chemical reactions taking place at the interface between core melt and concrete may generate various amounts of gases. When the quantity of gases is high, induced bubbles tend to favour a distribution of heat flux in an isotropic way. Then, the cavity in the concrete removed by the corium melt ablation is hemispherical (half ball-shaped): ablation depths in radial and axial direction being similar. On the other hand, when the quantity of gas is quite low, the distribution of heat flux is not isotropic. In this case, the radial direction is favoured by the natural heat transport processes through convection. The cavity ablation in the concrete is then more concave than spherical and tends to spread out with a horizontal radius getting larger than the vertical depth. These experiments have also shown values of radial/axial ablation ratios of between 3 and 4. I recognise that the complete understanding of the physical mechanisms involved in the ablation is yet to be developed.

183 EDF and AREVA have used the COSACO code in order to predict the behaviour of the MCCI process. The code is based on energy balance in which the decay heat is transferred to the interface between corium and concrete. But the model used within this code to transport energy assumes equal efficiencies to heat transfer to bottom and sidewall. COSACO therefore does not consider any anisotropic ablation with eventual higher heat fluxes to the side than to the bottom.

184 The composition of the sacrificial concrete used in the UK EPR reactor pit includes silica, not limestone; with silica, the amount of gas released during MCCI is expected to be rather low (higher with limestone). Hence, for the UK EPR, the distribution of energy is not expected to be isotropic: the ablation is expected to progress faster in the radial direction than axial. Furthermore, EDF and AREVA observed preferential radial ablation

in the FESICO test and have stated that they expect non isotropic erosion (TQ-EPR-1055, Ref. 9).

185 In summary, the methodology adopted within the code utilised to predict the MCCI progress does not include the capability to capture the anisotropic behaviour of the ablation process. However, the experimental test insights have demonstrated that the expected shape of erosion of EPR sacrificial concrete is likely to follow an anisotropic ablation progress. I consider the use of the adopted approach may produce uncertainties in predicting the ablation behaviour, leading to results that may not be adequately representative. An increased residence time is likely to have unintended consequences, such as, increased radiation duration. I have therefore raised the following Assessment Finding requiring justification that the current results are not invalidated by the uncertainties introduced by the 1D calculation methodology.

Finding

AF-UKEPR-CSA-12 *- The licensee shall, prior to construction – nuclear island safety-related concrete, provide an updated computational methodology to predict the MCCI progress within the reactor pit with a model of non isotropic ablation, supported by appropriate validation. This analysis could be performed by employing the existing COSACO model with different radial and axial heat flux efficiencies using values obtained from the 2D MCCI tests results.*

Protective Layer and Corium Interaction in the Reactor Pit

186 The reactor pit is covered by a protective layer, beneath the sacrificial concrete layer, expected to limit the progression of the core melt. This protective layer is made of ceramic bricks of ZrO_2 (zirconia) to withstand the high temperature of the core melt (circa 2500°C), offering a high thermal resistance. Potential interaction between corium and zirconia is not intended to cause any damage to the protective layer.

187 The radial erosion may be faster than the axial: in that case, the contact between oxide melt and protective layer will be extended. The average dissolution rate of the protective layer noted during COMAS tests analysing the corium spreading is less than 0.5 mm/min. Thus, the proposed protective layer thickness of 20 cm for EPR is adequate to avoid any significant damage before the molten core flows out of the reactor pit into the transfer channel.

188 During the period of melt retention within the reactor pit, due to the thickness of the EPR protective layer, the heat released from the corium should not significantly penetrate into the structural concrete behind, thus minimising any significant threat to the integrity of the protective layer. Moreover, EDF and AREVA have provided information to justify the structural stability with a postulated unexpected erosion thickness of 10 cm of the structure concrete (TQ-EPR-1069, Ref. 9).

189 I also note that STUK have performed an independent assessment of the transient heat conduction through the ZrO_2 ceramics. This assessment showed that the resulting temperature increase from the presence of corium and MCCI does not significantly penetrate the structural concrete. These analyses employ conservative assumptions and conclude that overall, for the period in which the melt is expected to remain within the reactor pit, the ZrO_2 provides thermal protection to the underlying structural concrete with a wide margin.

190 In summary, based on the evidence provided and the result of the STUK confirmatory analysis, I consider that the arguments relating to that the stability of the protective layer in the reactor pit has been appropriately addressed.

Pool Temperature and Liquidus Temperature

191 The spreading of the corium is highly dependent on the corium viscosity, which depends on the pool average temperature and the solidus-liquidus range of the mixture.

192 The PCSR states that the temperature of the oxidic melt within the reactor pit generally follows the evolution of the liquidus temperature. However, international research (Ref. 24) is not currently supporting of this position. In addition, it is noted that the arguments presented within the PCSR consider only 1D experiments in support of the adopted approach. Furthermore, it is also noted that the 2D OECD-MCCI experiments at Argonne National laboratory (ANL) have shown that longer term pool temperatures remain much lower than the liquidus temperature.

193 Given that, in cases where the temperature of the corium is lower, the expected solid fraction within the melt is higher and consequently viscosity is higher. I therefore believe that the pool temperature calculated by COSACO should be considered as having large uncertainties. I also note that the values of viscosity presented in the PCSR correspond to the higher range of temperatures leading to a lower solid fraction. I consider that the viscosity values assumed within the analysis are inappropriate.

194 However, the confirmatory analysis (Ref. 23), using high values of viscosity, has shown that the spreading process was satisfactory in most conditions examined. I therefore consider that the PCSR should be updated to reflect an analysis that is based on appropriate parameters. This has resulted in raising an Assessment Finding **(AF-UKEPR-CSA-20)** which requires an updated spreading analysis which includes appropriate values for the melt viscosity.

Reactor Pit Collection Effectiveness

195 The core catcher design concept considers, in a coupled approach, the energy required to ablate the sacrificial concrete, and the energy radiated from the surface of molten debris residing within the reactor pit to the remaining solid structures and components. The latter provokes further melting of the RPV bottom head leading to the second pour. The design intent is for this occurrence to take place prior to the melt plug failure. The supporting calculations have been performed using the COSACO code.

196 In the likely case of anisotropic ablation, where the lateral degradation is favoured, the core melt debris is expected to reach the protective layer on the side walls earlier. In this case, the lateral interface between the corium and zirconia bricks no longer represents a strong heat sink due to the low thermal conductivity of this protective layer. This will lead to a temperature rise within the molten pool, and increase the heat flux on the bottom and top surfaces. Such conditions enhance the axial ablation velocity in direction to the melt plug but also the heat radiated to the RPV, leading to an earlier melt plug failure but also to an earlier second pour of remaining melt from the RPV. Thus it is plausible that the conditions that may lead to a second pour prior to the melt plug failure may still be present.

197 In summary, I have assessed the effectiveness of the reactor pit in retaining the molten material likely to be released from the RPV so that it can be conditioned prior to its continuous release via the melt plug opening/failure. I consider that the reactor pit is

likely to meet its safety requirements subject to confirmation of the revised calculations using an updated model in COSACO to capture anisotropic ablation. This requirement has been identified by the Assessment Finding **AF-UKEPR-CSA-12**.

Uncertainties on the Amount of Concrete Ablated

198 The PCSR describes a layered configuration of corium within the reactor pit. The PCSR also examines the potential for a mixed configuration, for which overall core melt properties are mainly driven by the oxidic fraction. In layered configuration, the density of the concrete slag layer is lower than the metallic layer, which in turn is lower than the oxidic layer. The PCSR states that due to the continuous absorption of lighter components from the concrete, the oxidic layer density is expected to become lower than the metallic layer density, leading to the inversion between the layers: metallic on the bottom and oxidic on top. The PCSR indicates that this is expected to occur shortly before the end of the MCCI in the reactor pit.

199 The PCSR takes credit for the layer inversion in order to reduce the probability of a steam explosion: thermal conductivity of the metallic components is several times higher than the oxidic layer, thus interactions with any water present would be several times more energetic, by potentially producing very large quantities of steam.

200 The mass of ablated concrete within the reactor pit has been the subject of a number of technical queries (TQ-EPR-1061, 1066 and 1328, Ref. 9) during GDA Step 4 assessment. In response, EDF and AREVA have provided the mass of concrete likely to be affected within the reactor pit for the expected corium height. In assessment of these, it became apparent that inconsistencies in these responses are due to different modelling approaches and assumptions employed.

201 The uncertainties resulting from the adopted approach could lead to an additional mass of ablated concrete of circa 20%. In the case of the ablated mass being underestimated, the response provided regarding the robustness of the layer inversion is acceptable. It is however necessary to justify that the resultant corium viscosity has not been significantly affected. In the case of the ablated mass being overestimated, it is necessary to demonstrate the robustness of the layer inversion phenomenon.

202 In summary, considering that the mass of ablated concrete is one of the key factors affecting the corium viscosity influencing the spreading capability and potentially the layer inversion; I have raised the following Assessment Finding.

Finding

AF-UKEPR-CSA-13 *- The licensee shall, prior to construction – nuclear island safety-related concrete, demonstrate the presence of the layer inversion phenomenon for the bounding scenario of the minimum ablated concrete quantity. This justification is required to ensure that the risk associated with any significant interactions between water and the metallic layer is avoided. The response should also demonstrate that the resultant corium viscosity is appropriate for the bounding scenario of the maximum ablated concrete quantity.*

4.3.5.3 Melt Plug Operations (Failure / Opening)

203 In order to perform maintenance activities within the reactor pit, the design incorporates access from the spreading area, through the melt plug located at the bottom of this

space. The melt plug can be removed and is equipped with dedicated components such as a locking mechanism, and an inner and an outer frame. These features have no specific role during accident progression nor contribute to the plants safety objectives in case of accident scenarios.

204 The melt plug is covered with the same thickness of sacrificial concrete as the reactor pit. The concrete layer is supported by a metallic structure including an aluminium plate, called the “gate”. The aluminium has a low melting point, thus, after erosion of concrete the gate will melt. This could lead to either a partial or total failure of the plug and will result in the flow of corium into the transfer channel and subsequent relocation into the spreading room.

205 The safety objectives of the melt plug are:

- withstand an adequate duration to allow the melt collection, retention and conditioning of the corium within the reactor pit, and
- provide adequate cross sectional area to enable the corium to discharge continuously and effectively into the transfer channel in a viscous state.

Assessment

Mechanical Resistance of the Melt Plug

206 To permit removal of the plug for maintenance and inspection, there is no permanent fastening mechanism between the gate and the surrounding structural concrete. An early mechanical failure of the melt plug resulting from the impact of corium, needs to be addressed.

207 In response to queries (TQ-EPR-1064 and 1239, Ref. 9) regarding the structural justification of the melt plug frames and locking mechanism, EDF and AREVA provided additional design information regarding the locking system, the outer/inner frame and the location of the welds. This included stress calculations for the locking bolts and for some of the welds. I recognise that the outer frame includes 88 welds and for the purpose of stress calculation, these can effectively be grouped in three or four potential subsets. I therefore requested the stress calculations for each potential subset and the outer frame (housings of the locking bolts). Although a response has been provided, I do not consider this to be a comprehensive response.

208 Manufacturing defects lead to local stress concentration. The acceptability of these defects depends upon the ratio between the calculated stress and the allowable stress indicating the margins required to account for any potential defects. For the locking bolts, this ratio appears to be quite low and I am satisfied that the margin is thus relatively high.

209 On the other hand, the calculated ratio for the welds is high leading to a reduced tolerance for potential manufacturing imperfections. I thus requested further information regarding the design intent to eliminate the risk of unacceptable manufacturing defects (TQ-EPR-1367, Ref.9).

210 EDF and AREVA in their response stated that “*all weld seams of the melt plug and Support Frame will be inspected by visual examination only. Quality level C according to EN ISO 5817* (Ref. 17) *applies for evaluation of the welds*”. The response also stated that the allowable stress is systematically set at 60% of the yield strength by applying the KTA 3205.2 standard (Ref. 18). I accept that applying the German KTA 3205.2 standard allows an additional margin of 40%. However, this fixed value results from implementation of this standard and not from a dedicated risk analysis, as is required in

the UK. The application of these standards is not immediately apparent in a UK perspective and EDF and AREVA are requested to provide a satisfactory justification for the adequacy of the 40% margin.

211 The EN ISO 5817 standard identifies three quality levels for finished welds, B, C and D; where Level B corresponds to the highest requirement. EDF and AREVA in the response to TQ-EPR-1367 (Ref. 9) have not presented the rationale for selecting the quality Level C for these welds. In addition, in EN ISO 5817 the quality levels refer to production quality and not to the fitness-for-purpose of the manufactured product. I therefore believe that EDF and AREVA should provide additional justification for the selected quality level, although the adequacy of the mechanical integrity of such components falls within the assessment remit of other disciplines within the ND.

212 EDF and AREVA's response states that only visual examination of these welds is required (TQ-EPR-1367, Ref. 9). However, the quality Level C of the EN ISO 5817 includes many criteria for internal imperfections; such as cracks, types of porosities and lack of fusion. Given that internal weld imperfections can not be detected by visual testing, it is therefore considered that quality Level C criteria cannot be met by visual examination only.

Finding

AF-UKEPR-CSA-14 *- The licensee shall, prior to construction – nuclear island safety-related concrete, provide additional justification to:*

- *demonstrate that the weld beads and outer frame meet the loading requirement, and*
- *support a testing programme to capture unacceptable defects in the weld beads*.

Creep Hazard of the Welds During MCCI

213 During ablation of the concrete adjacent to the melt plug, the outer frame welds will be heated by conduction from the metallic parts that are in contact with the core melt. This could cause a significant degradation of the weld mechanical properties leading to a potential premature collapse of the melt plug.

214 In a response to an enquiry, TQ-EPR-1367 (Ref. 9), relating to creep hazards, EDF and AREVA have provided the results of calculations predicting the weld temperatures. These calculations included two cases; i.e. a fast and slow rate of ablation. In the scenario of slow ablation rate (considered to be more onerous than fast ablation) the results show that the temperature at the weld is less than 200°C. Such a temperature offers a sufficiently high margin from the level that would adversely impact on the stability of the weld material. I therefore have considered the response to be adequate.

215 Additionally, in the response regarding the mitigation of an early mechanical failure (TQ-EPR-1367 and 1239, Ref. 9), EDF and AREVA have emphasized the presence of two M16 screws. They have stated that "*In case of a postulated weakening of the dedicated measures that keep the support frame in place during melt accumulation, the presence of these screws is expected to let the entire frame become jammed within the enclosure provided by the zirconia.*" EDF and AREVA have explained the role of these M16 screws: they are used to support the installation of the outer frame on the ceiling of the transfer channel. In considering the response, I have a reservation about the claim made relating to the functionality of these screws in severe accident conditions. I judge

that the melt plug including the corium present in accident conditions is retained in position by the locking features and the frames, and that no safety function can be placed on these screws.

Two-Pour Scenario – Impact of Bumpers

216 The bumpers are located within the reactor pit at some distance from the melt plug. In case of an unzipping of the RPV in a one-pour failure scenario, these are designed to protect the melt plug from the resulting shock. These bumpers are made from the same sacrificial concrete as that in the reactor pit, and are expected to be eroded in a similar manner.

217 As discussed in Section 4.3.3.1, previous experiments have shown that the ablation rate in the radial direction is expected to be in the order of 3 times more effective than in the axial direction. The PCSR states that the second pour is expected to occur when the corium from the first pour has eroded roughly 20 cm of the sacrificial concrete layer in the axial direction. This will lead to a likely radial (horizontally) ablated thickness in the order of 60 cm, being more than half the bumpers width. Given the nominal width of the bumper is 90 cm, an ablated thickness of 45 cm on each side wall will lead to the collapse of the bumpers into the pool of corium.

218 In response to a query (TQ-EPR-1066, Ref. 9) relating to the potential interference of collapsed bumpers with melt plug erosion, EDF and AREVA stated that concrete chunks resulting from any collapsed bumpers would not influence the melt plug opening because they are expected to dissolve while floating on the pool surface.

219 Although I concur with the statement that concrete chunks will float on the pool, I have assessed the consequences of the RPV bottom head failure and its subsequent drop onto the bumpers in two-pour accident scenarios. In such scenarios, the concrete chunks or degraded bumpers could be pushed directly through the liquid pool onto the top of the melt plug. This potential configuration is not identified by the current UK EPR PCSR and may lead to consequences on melt plug failure that has not been examined within the safety submission.

220 I have therefore raised an Assessment Finding requesting clarification to justify the failure of the melt plug for two-pour scenarios where the toppled bumpers are likely to impact directly on the melt plug transmitting the load from the fully laden RPV bottom head. In addition, the safety submission should include conditions where the presence of concrete blocks trapped by the RPV lower head could prevent a uniform melt pool being present on the surface of the plug.

Finding

> ***AF-UKEPR-CSA-15*** *- The licensee shall, prior to construction – nuclear island safety-related concrete, justify that potential presence of chunks of concrete above the melt plug at the time of bottom head failure has no significant consequences on the melt plug opening.*

In-service Maintenance

221 The frequency of maintenance operations in the reactor pit is not clearly defined for the UK EPR. EDF and AREVA anticipate that the reactor pit will be regularly inspected which should provide some confidence regarding the melt plug condition. It can however be

assumed that in between inspection periods, water may accumulate on the concrete of the gate, being the lowest level within the reactor pit. This could potentially, in the longer term, degrade the quality of the concrete casting and the reinforcement in this sacrificial concrete. The impact of such degradation is difficult to quantify in terms of plant assurance over 60 years. This may not directly impact on the core catcher process on demand but could impact on the requirement to be able to withstand an earthquake, although the adequacy of the seismic qualification falls within the assessment remit of Civil Engineering disciplines of the ND which is reported in Ref. 28.

222 In their response to a query (TQ-EPR-1062, Ref. 9), EDF and AREVA have stated that the inner melt plug frame, the locking mechanism below the gate and the outer frame were made of stainless steel. Thus, those metallic parts should not exhibit any significant degradation over time. More clarity is required addressing the specific maintenance requirements and impact of any water ingress on concrete stability.

Finding

> ***AF-UKEPR-CSA-16*** *- The licensee shall, prior to inactive commissioning – containment pressure test, define the examination, maintenance, inspection and testing requirements necessary for the melt plug to fulfil its safety functions.*

4.3.5.4 Flow of Corium into the Spreading Compartment

223 The UK EPR design includes a spreading area located off-centre under the RPV. The spreading compartment is the final destination of the molten debris discharged from the reactor pit via the melt plug and transfer channel. This allows for cooling of the corium by increasing its surface to volume ratio and, by introduction of controlled water flow, providing long term stabilisation of the melt. The spreading process has the following safety objectives:

- transfer of the molten material in a single continuous flow,
- completion of the transfer prior to flooding of the spreading compartment,
- even distribution of the molten material within the spreading area, and
- availability of a large surface-area-to-volume ratio to ensure the production of a solid surface crust.

Assessment

Long Term Protective Layer Stability

224 In response to an enquiry (TQ-EPR-1346, Ref. 9) relating to the long term stability of zirconia under normal operating conditions, EDF and AREVA stated that the corrosive hazard from humidity is eliminated by the protective design features. These features include a layer of sacrificial concrete covering the zirconia layer within the reactor pit, and a coated steel liner over the zirconia layer within the transfer channel. EDF and AREVA have also provided a document (Ref. 19) clarifying the long-term stability of zirconia against neutron irradiation, considering higher flux levels than those expected during normal operating conditions.

225 However, this response does not provide any justification for the sufficiency of concrete and steel layers to prevent humidity ingress over a 60-year lifespan, and whether it is

desirable to inspect the actual stability of the protective layer, in the frame of a surveillance programme, in order to confirm that there is no long-term significant deterioration of the zirconia. I therefore consider that a potential deterioration during the prolonged lifespan may result in rapid loss of the protective layer functionality in the case of a severe accident.

Finding

> ***AF-UKEPR-CSA-17*** *- The licensee shall, prior to construction – nuclear island safety-related concrete, provide the surveillance programme to monitor the zirconia stability for the plant lifetime*.

Protective Layer and Corium Interaction in the Transfer Channel

226 The floor, ceiling and sidewalls of the transfer channel are covered by layers of zirconia (ZrO_2) and coated steel. The schematic view of this feature is illustrated in Figure 3. In this channel, the governing interactions between corium and zirconia are different to those within the reactor pit such that:

- Failure of the melt plug introduces the corium into the channel which rapidly melts the steel cover, subjecting the zirconia to a thermal shock. This may potentially damage the zirconia bricks.
- On completion of melt transfer, a residual mass of corium remains indefinitely within the transfer channel potentially interacting with the zirconia protective layer.

227 In response to an enquiry (TQ-EPR-1056, Ref. 9) relating to the risk of interactions with the protective layer within the transfer channel, EDF and AREVA provided additional explanation about the possible interaction mechanisms between corium and zirconia:

- *Thermal shock* - Results from experiments involving pouring of superheated iron/alumina melt over zirconia bricks have shown that the surface layer of bricks remain generally intact once the molten material is cooled down, with only a few cracks appearing on the side walls. As a result, I am content with the argument that the protective layer can withstand the thermal shock.
- *Residual corium fraction within the transfer channel* - This residual material is either frozen or it has exhausted its momentum at the end of the outflow from the reactor pit. EDF and AREVA anticipate that due to the temperature gradients, a thin layer of re-solidified material will rapidly develop covering the surfaces within the channel. EDF and AREVA have estimated that an average thickness of ~5 cm may cover the entire floor area of the transfer channel, corresponding to a mass of ~3 metric tons.

228 Given the introduction of passive cooling to the spreading compartment is planned to occur shortly after the spreading of corium is complete; this water is likely to flow into the transfer channel. This will quench the residual corium by a variety of heat transfer mechanisms irrespective of its composition.

229 EDF and AREVA have provided an estimate that residual corium remaining within the transfer channel would need to have a minimum thickness of approximately 10 to 14 cm to prevent it from a short term re-solidification. This is greater than the predicted 5 cm height of residual corium within the transfer channel.

230 In summary, I consider that the interaction between zirconia and residual corium in the transfer channel has been examined within the safety submission and the depth of the

residual corium will allow re-solidification without any significant degradation of the zirconia.

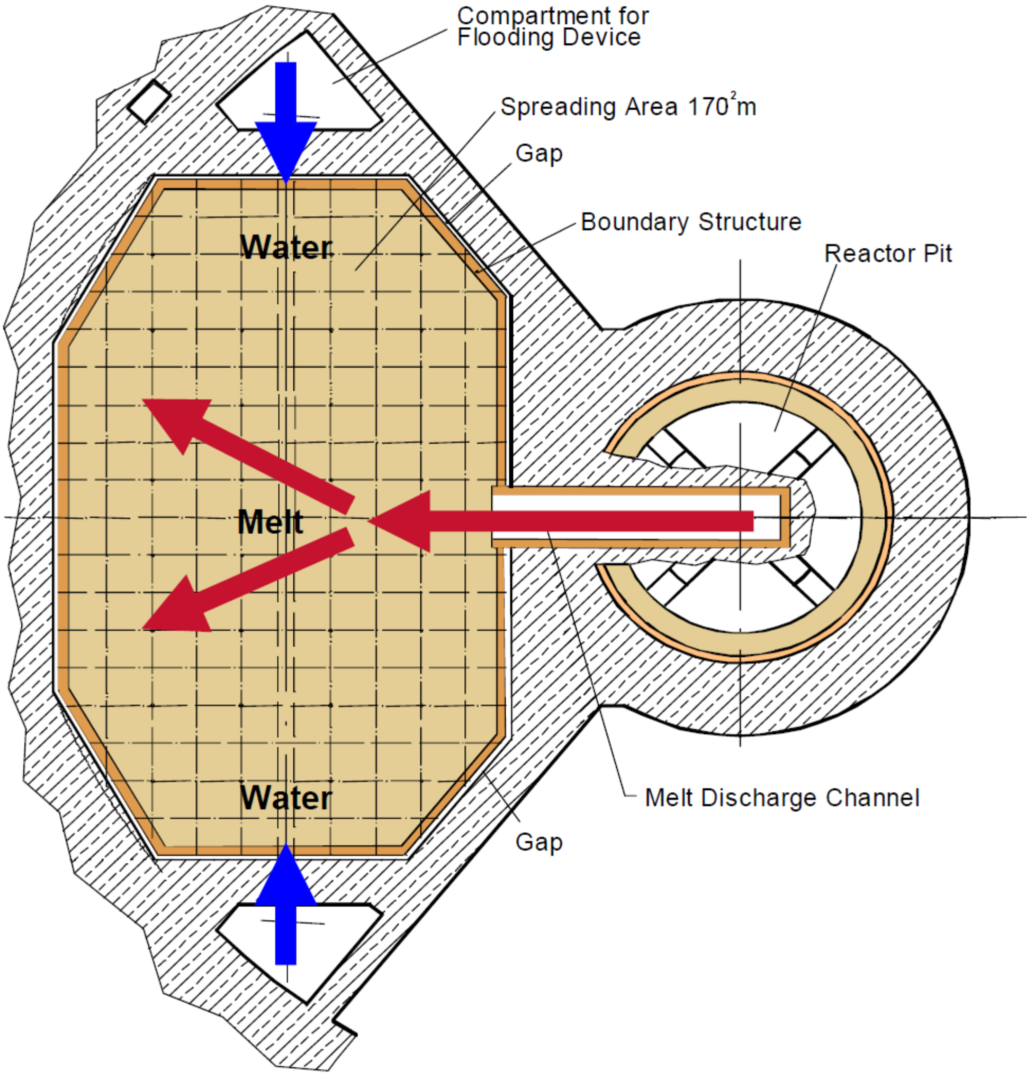

Figure 3: Schematic View of the Transfer Channel and the Spreading Area

<u>Spurious Activation of Flooding Valves</u>

231 The presence of significant quantities of water within the spreading compartment in severe accident condition will adversely influence the pressure peaks within the containment due to vaporisation on melt arrival and increases the likelihood of steam explosion, which is further discussed in Section 4.4.

232 The safety objectives of the initial cooling are that the pressure peaks due to steam generation from flooding and quenching of the upper melt surface should not pose any threat to containment integrity. EDF and AREVA in response to my enquiry (TQ-EPR-1062, Ref. 9) relating to the potential presence of water in the spreading compartment and its detection, have stated that a sensor will be installed in the spreading compartment

to detect water and to send an alarm to the Main Control Room in order to shutdown the plant. I am satisfied that this meets the requirements of SAP FA.16 requiring consideration of severe accidents.

233 Not withstanding the ability to detect water in the core spreading compartment, it is still conceivable that the IRWST water level could fall over a period of time. It is therefore necessary for IRWST to ensure that adequate water level is maintained within the tank. This can be ensured by addition of instrumentation or surveillance and I have raised an Assessment Finding in order that this issue could be addressed.

Finding

> ***AF-UKEPR-CSA-18*** *- The licensee shall, prior to construction – nuclear island safety-related concrete, justify that suitable arrangements are in place to ensure that the IRWST water level is adequate for reasonably foreseeable faults*.

4.3.5.5 IRWST Passive Injection into the Spreading Compartment

234 The PCSR claims that there is no inflow of water from sprays or leaks and only a limited amount of condensate could form inside the spreading compartment. However, significant accumulation of water quantities within the spreading compartment may lead to an increased risk of steam explosion on arrival of the corium into this area. The assessment of steam explosion is covered as a dedicated topic and described in Section 4.4 and it will not therefore be further discussed in this section.

235 The water delivery from the IRWST to the core catcher is via two independent lines within the CHRS. The IRWST water injection is triggered by the thermal destruction of metallic receptors which relief pre-stressed steel cables linked to the passive flooding valves. The safety submission indicates that the time taken to fill the passages from these valves to the top of the melt, under gravity, is in the order of a few minutes to minimise local melt-through and excessive damage to the supporting structure. On each line, the design includes a motor-operated isolation valve (normally open) located upstream of the passive flooding valve. The safety objectives of the IRWST injection into spreading room are:

- delayed IRWST water injection onto the molten pool for the prescribed duration to allow completion of the spreading process, and
- flooding the molten pool to promote superficial fragmentation to improve coolability.

Assessment

<u>Passive Outflow Reducer</u>

236 The Passive Outflow Reducer (POR), located on the line between IRWST and spreading compartment, is required to prevent the flow of water into the IRWST during active cooling mode, when water circulation is driven by CHRS pump. The POR is a novel design feature which replaces the use of a non-return valve and offers a higher reliability in severe accident conditions.

237 In response to an enquiry relating to the margins for resistances influencing the pressure drop in forward and reverse flows for bounding scenarios (TQ-EPR-1065, Ref. 9), EDF and AREVA have explained that the POR does not include any internal moving parts, and thus the flow resistance offered by this device is dominated by its shape.

238 The low flow resistance of the POR in the forward direction will favour the gravity-driven flooding of the spreading room from the IRWST. EDF and AREVA have demonstrated that the spreading room is adequately flooded when water flow from the IRWST is affected due to reduced hydrostatic pressure and a pressure increase in the spreading room due to steam generation. EDF and AREVA have also provided calculations to justify that the backward flow resistance includes sufficient margins to prevent the flow into the IRWST for the bounding (lowest) water level within the IRWST. This justification for the reverse flow resistance is relevant to the active cooling mode.

239 I have considered the response and I am content with the justification provided.

Passive and Isolation Valves

240 At the early stages of a severe accident, the IRWST water injection into the spreading compartment is via two independent trains, each incorporating a passive valve initiated by the thermal destruction of metallic receptors which relief pre-stressed steel cables linked to the passive flooding valves. The water injection line includes a dedicated “leak recovery tank” that is housed within the same room as the passive and isolation valves. The dedicated tank houses a water detection sensor which initiates an alarm for the operator. In case of a leaking passive valve, the operator would have the option of using the isolation valves to isolate the line.

241 The closure of the isolation valve will prevent the discharge of the IRWST water on demand. In addition, the successful opening of the passive valves could also be hindered by fouling of these valves, severely restricting the flow into the spreading compartment on demand.

242 In response to queries relating to the maintenance of these valves (TQ-EPR-1062 and 1346, Ref. 9), EDF and AREVA have stated that;

- Tests will be performed periodically to verify the functionality of these valves.
- In the case of leakage rate higher than 1000 l/year from one valve, the injection line is isolated; and, for leakage rate lower than 1000 l/year the water is stored in a corresponding dedicated tank.

243 There are only two cases when an isolation valve may be closed: either there is maintenance on the passive valve on the same line or this passive valve has a leak which is with a rate higher than 1000 l/year. Water leakage greater than 1000 l/year from the passive valves will lead to the closure of both isolation valves; consequently EDF and AREVA have stated the reactor will be shutdown in order to repair the leaks.

244 In summary, the need for maintenance on the passive valves and mitigation measures for any potential leakage has been addressed.

4.3.5.6 Initial Corium Cooling due to Flooding

245 The introduction of IRWST water onto the corium within the spreading compartment and quenching of the upper melt surface will lead to a significant generation of steam that will be released into the containment. This release of steam will cause a major pressure peak within the containment that may pose a threat to its integrity. The safety objective is to ensure such pressure increases will remain within the containment design pressure limits.

Assessment

246 EDF and AREVA have provided the methodology and the results of the calculations predicting the containment pressure transient relating to the initial cooling of corium during the passive mode. The IRWST supply lines are expected to supply water at a rate of approximately 45 kg/s each (~90 kg/s in total). EDF and AREVA have used an initial heat flux of 3 MW/m^2 on top of the core melt, which corresponds to the value observed during the MACE experiment. It is however recognised that in reality the heat flux will decrease as the transient progress.

247 EDF and AREVA have predicted that during the early part of the flooding, all the water introduced is converted into steam and is unlikely to fully cover the corium surface in its totality. This minimises the risk of steam explosion due to the absence of reasonable depth of water necessary to create the required conditions for steam explosion. These calculations assume that the temperature of the fragmented oxidic layer (crust) has reached the water saturation temperature, and that the bulk corium pool temperature has reached the immobilization temperature (this is defined as the pool temperature when the fraction of the solidified material reaches 50% within the pool). These calculations neglect the metallic layer (interacting with the concrete layer below it) that would normally act as a heat sink to the oxidic layer. The result of this analysis indicates a maximum containment pressure of 4.9 bars representing a margin to that of the maximum containment design pressure.

248 In summary, I have assessed the assumptions used in the analysis to predict the heat transfer between water and corium within the spreading compartment during the early part of the transient. These assumptions tend to result in a higher heat transfer into the water and higher steam generation, and thus lead to higher than expected containment pressure that even so does not challenge the maximum design conditions.

4.3.5.7 Long Term Cooling of the Melt in the Spreading Compartment

249 At the end of spreading, the corium is expected to be contained within the engineered cooling structure in the spreading compartment and ultimately re-solidify. The long term cooling of the melt calls on both CHRS trains. Each train is capable of meeting the expected cooling requirements, and is intended to operate for 12 months before maintenance is required.

250 The safety objectives of the long term cooling are the following:

- Corium is safely enclosed within the spreading compartment, long-term, allowing the containment liner and basement integrity to be preserved.
- Decay heat is removed from the melt, long term.

Assessment

Cooling plates capability

251 When MCCI ends in the spreading compartment, core melt is then in contact with metallic cooling plates which are flooded underneath with water from the IRWST. Core melt is then sandwiched between water on the top side, and cast iron water-cooled plates at bottom side. In order to fulfil their cooling function, the cooling plates must not degrade significantly.

252 In a response to an query relating to the degradation of the cooling plates, (TQ-EPR-1350, Ref. 9), EDF and AREVA have confirmed that the expected scenario of stratified

corium with a metallic layer at the bottom, is the most likely case leading to highest heating of cooling plates. For this scenario, the maximum temperature reached by the plates is approximately 700°C, which remains below the melting point of iron.

253 The EDF and AREVA calculations do not appear to reflect the transient occurrences of rapid hot spots (greater than 800°C in the upper side of the cooling plates) as observed during the OECD-MCCI2 WCB1 experiment that employed cooling plates thinner than those proposed for the EPR design. To progress with this experiment, it was decided to reduce the input power to the melt by 50% to prevent further temperature rise within the plates. However, I have discussed this issue with STUK. They have carried out experiments and demonstrated that although steady heat transfer analysis would predict local dryout on parts of the plate, the presence of "intermittent plug flow" regime ensures adequate heat removal by a process of drying and quenching. I therefore conclude that the cooling plate design is likely to function satisfactorily.

254 The spreading plates incorporate fins which provide cooling channels to direct water flow to cool the bottom side of the corium.

255 In a response to a query (TQ-EPR-1062, Ref. 9) relating to the possibility of cooling channel plugging by potential debris that may impact on the overall performance of the system, EDF and AREVA have indicated that these channels are not accessible after construction is complete. EDF and AREVA are therefore planning to check the status of these channels during construction.

256 I considered the case of a blockage of the cooling channels under the spreading plate. EDF and AREVA state that such a blockage would induce a slow melting of the top of the plate, but they have not provided the minimum value of the blocked "*larger area*" from which the entire melting of one or several plates is possible. They have not examined the stability of the blockage in order to determine whether this phenomenon would tend to stop or to spread. I have therefore made a finding requiring that this be addressed.

Finding

> ***AF-UKEPR-CSA-19*** *- The licensee shall, prior to construction – nuclear island safety-related concrete, demonstrate the level of blockage in the channels, under the cooling plates, that can be tolerated before their safety function is impaired*.

Use of CHRS Trains for Long-term Cooling

257 The distribution of energy during the long-tem cooling is a key aspect of the severe accident management strategy. The main system used by the operator is the Containment Heat Removal System (CHRS), equipped with two trains. This system has two main functions: provision of the spray within the containment to reduce the pressure, and supply of water to the core catcher in the active cooling mode. In addition, the strainers back-flushing function also utilises the same system. The ultimate heat sink is through a heat exchanger on each train located externally to the containment.

258 In response to an enquiry (TQ-EPR-1313, Ref. 9) relating to the overall energy distribution within the containment and the management of the CHRS operational modes, EDF and AREVA have provided further additional information that active cooling mode is used to avoid generation of saturated water in the spreading compartment limiting fission product release into the containment atmosphere.

259 EDF and AREVA claim that if one CHRS train is used the water temperature is lower than the saturation temperature.

260 I have accepted the assurance given by EDF and AREVA that each CHRS train can perform the duties expected from this system, and would therefore judge that the requirements of SAP FA.16, requiring the severe accident mitigations, have been met.

4.3.6 Confirmatory Analyses – Severe Accident Progression

261 To examine the claims made within the UK EPR PCSR for the containment thermal hydraulics response and the performance of the severe accident mitigation features, I commissioned Sandia National Laboratories (SNL) to perform a set of confirmatory analyses. SNL has used the MELCOR severe accident analysis code to examine the UK EPR containment and severe accident performance for a number of bounding scenarios. These will demonstrate severe accident management strategies inherent in the reactor plant's design. Two key aspects of the UK EPR design are the management and control of hydrogen produced in a severe accident, and the long-term stabilisation of molten core materials. The containment's integrity must be assured in the face of these predicted demands from sources such as hydrogen combustion and steam pressurisation.

262 The UK EPR design makes use of dedicated containment design features that are intended to promote good global transport and mixing of steam and hydrogen produced in accident conditions. The use of passive hydrogen recombiners distributed throughout the containment will help to convert hydrogen back to steam. Both effective gas mixing and steady recombination of the hydrogen gas is needed to ensure detonable mixtures of hydrogen and air which can threaten containment integrity, do not accumulate locally or globally. Effective functioning of these features greatly minimises this traditional threat to containment integrity in ways significantly improved over earlier reactor containment designs. MELCOR confirmatory analyses are also intended to evaluate the performance of these hydrogen control features and explore the implications of partial non-functionality of the hydrogen recombiners.

4.3.6.1 MELCOR Confirmatory Analysis

263 The UK EPR design also includes a primary system depressurisation strategy that has two principal aims. The first aim is to assure system depressurisation so as to enable accumulator injection into the core and subsequent activation of a low pressure coolant injection system. The other aim of the RCS depressurisation is to avoid any possibility of high pressure melt ejection and its consequences including the potential for so-called Direct Containment Heating phenomena which can lead to early containment failure. Successful activation of the depressurization system with successful low pressure injection should avert core damage. MELCOR confirmatory analyses investigate the implications of failure of this important and principal design feature and show ultimate accident recovery even with failure of the PDS, provided low pressure injection is available. Failure of the hot leg may result if PDS fails.

264 Long term melt stabilization from a core melt accident is a traditional safety concern with severe accidents. Molten core materials may fail the reactor lower vessel head and come in contact with concrete in the reactor pit and spreading area. MELCOR has the capability to model MCCI. There will be steady accumulation of non-condensable gases which may produce containment failure by static overpressure unless containment venting procedures are accomplished.

265 The MELCOR confirmatory analyses are extended from the traditional PWR basemat modelling to evaluate the CMSS engineered spreading strategy and confirm the success of the strategy.

266 MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in PWRs. MELCOR is under ongoing development as an advanced plant risk assessment tool at Sandia National Laboratories for the US NRC. A broad spectrum of severe accident phenomena in PWRs is treated in MELCOR in a unified framework. These include thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heat up, degradation, and relocation; core-concrete attack; hydrogen production, transport, and combustion; fission product release and transport behaviour.

267 In development of MELCOR SNL continues to receive significant developmental support from the US NRC, and through the CSARP International research cooperative. In recent years, MELCOR development activities have focused on implementing best-knowledge modelling of core melt progression processes within the core region, the lower vessel head and core catchers. The modelling development is based on the body of research around Phebus, MASCA and other international research programmes including improved modelling of fission product speciation, release and transport based on Phebus and Vercors testing programmes. A MELCOR validation document exists (Ref. 38).

268 The MELCOR model used in the confirmatory analyses is quite similar in terms of nodalisation with the analogous MAAP model used by EDF and AREVA. The original MELCOR model was developed by ERI for the US NRC for US design certification activities. The model was subsequently obtained by SNL and updated for use with the latest MELCOR code version (MELCOR 1.8.6) and the control system was generalised to allow for the examination of a wide variety of accident sequences. Similar to the MAAP model, the MELCOR model used 5 radial core rings and 12 axial core levels. Hydraulic nodalisation allowed modelling of 2-D in-vessel natural circulation and special hot leg nodalisation captured important counter current natural circulation phenomena.

269 Creep rupture modelling is used to monitor for and allow for RCS failure of the hot leg nozzle, steam generator tubes and the vessel lower head. PARS are distributed throughout the containment to control hydrogen accumulation, facilitated by the passive opening of rupture foils and dampers that encourage hydrogen mixing. Finally, an approximate model for the reactor pit, sacrificial concrete and melt plug and for the spreading room floor and associated water cooling system was implemented in the MELCOR input deck. The MELCOR model made use of the CORCON MCCI modelling and represented accurately the specific concrete compositions of the EPR pit and spreading room area.

270 The severe accident sequences modelled in the MELCOR UK EPR confirmatory analyses were all variations around the Station Blackout (SBO) accident scenario. The basic SBO involved immediate loss of feedwater and four leaking pump seals. The base case analysis assumed successful activation of the PDS. This analysis compared very well with the analogous RP supplied MAAP analysis. There were good comparisons of steam generator dry-out time, core water level behaviour, total hydrogen produced and overall containment response, including successful demonstration of the ex-vessel reactor pit and spreading compartment material stabilisation. Additionally, virtually all of the hydrogen generated in the accident was converted to steam by the PAR recombiners simulated by SNL, Ref. 22.

271 Variants on the base case SBO were explored subsequently where the PDS depressurisation system was assumed to fail. In these cases, high temperature natural circulation in the RCS hot leg produced hot leg nozzle failure by creep rupture where upon RCS depressurisation occurred. Again, the main sequence signatures were quite comparable with MAAP analyses and following vessel lower head failure, core melt spreading, cooling and stabilisation were observed. Full hydrogen control was demonstrated by the PAR recombiners. A second variant on this PDS-failure sequence assumed activation of the low pressure injection system following RCS hot leg failure and depressurisation. This sequence demonstrated successful arrest of the accident progression by re-flooding of the core.

272 Finally, a last variation on the PDS-failure sequence assumed that significant numbers of the PAR hydrogen recombiners were inoperative, that containment sprays were activated but that activation of cavity flooding had failed. This final most challenging case produced essentially the same amount of hydrogen as other SBO cases, but only half of the hydrogen was effectively managed by the PAR recombiners, with the other half being consumed in hydrogen deflagrations. It is significant to note that predicted combustion of the hydrogen that was not recombined by the PAR's neither challenged the containment nor threatened to produce detonable mixtures, even with some degree of steam de-inerting produced by the containment spray system.

273 Generally, the MELCOR predicted results were largely consistent with the MAAP results presented by EDF and AREVA with respect to accident timing, mass of hydrogen produced in-vessel and CO/CO_2 produced ex-vessel, although small differences in steam generator dry-out time was observed. Ex-vessel melt spreading and stabilization was confirmed in the MELCOR analyses in all cases examined. Hydrogen control was demonstrated in all cases with 100% of PARS functioning. For the case with significant failed PARs, about half the hydrogen was controlled by the remaining functioning PARs, with the result that the remaining half of the un-reacted containment hydrogen was consumed in burn events (deflagrations), principally in the upper dome region. It is recognised that the choice of operational PARs and their locations is influential on the outcome of the predictions.

4.3.6.2 Confirmatory Analyses - Corium Flow Behaviour

274 The transfer of corium from the reactor pit to the spreading compartment is an important step in successful progression of the CMSS process. In order to examine the effectiveness of corium spreading from the melt plug to the spreading compartment, EDF and AREVA employed the CORFLOW code and a complementary analysis based on a phenomenological spreading model developed by the Royal Institute of Technology (RIT), Stockholm. The European Severe Accident Research Network, SARNET, in 2007 questioned the applicability of the simplified approach raising the following technical points with regard to the RIT model:

- A 2D solution has been used for estimating the viscous spreading velocity in a 1D channel. The result presents a mix of overestimated and underestimated parameters which introduces two sources of uncertainties.
- The large impact of the substratum heat transfer on the calculated spreading length.
- Square root averaging of two spreading lengths used in the correlations only provides an estimate.

275 In order to examine the claims made for spreading of the core melt within the spreading compartment, I commissioned Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) to perform a set of independent confirmatory analyses to develop an appreciation of the extent of the uncertainties using this independently developed methodology.

276 The alternative methodology is based on the following features;

- The cooling of the melt by radiation to adjacent structures leads to the formation of a crust with a thickness "s" as a function of time.
- The spreading of the melt leads to a reduction of the melt average depth.
- Heat transfer to the bottom surface and sidewalls is neglected.
- Empirical adaptation to results of spreading experiments has been used to define the stopping criterion.

277 To examine the spreading efficiency (the ratio between the actual area covered by the corium and the total available spreading area), GRS performed confirmatory analysis of three cases presented within the safety submission. These scenarios covered;

- case A – short pour with m_A= 332,000 kg and Δt_{inflow} = 30s;
- case B - medium pour with m_B= 350,000 kg and Δt_{inflow} = 90s; and
- a third case, called scenario1 in the supplementary analysis performed for AREVA – with m_C= 403,000 kg and Δt_{inflow} = 178s.

278 Where "m" represents the mass of the material and "Δt" is the duration of the pour.

279 The results of the confirmatory analysis shows that for cases A and B, the calculated spreading efficiency is higher than those presented within the safety case, confirming that all the inflow of corium could be spread within the available spreading area with a reasonable margin.

280 The result of the calculations for the third case also showed that the incoming corium can spread evenly within the spreading compartment, though with much reduced margins. This has shown that a significantly larger mass of molten material and a longer period of pour coupled with a relatively small opening cross section would still lead to a reasonable spreading over the available surface area. I also note that the PCSR argues the failure of the melt plug with relatively small opening is a low probability event.

281 GRS has provided an uncertainty analysis based on its model in which parameters have a large range of possible values. Notably, the viscosity spectrum spreads over about one order of magnitude; the maximum values exceed 1 Pa.s, though the maximum value considered in the safety submission is 0.034 Pa.s. The use of low viscosity values within the PCSR have already been noted regarding to the pool temperature and the liquidus.

282 Three samples with a spreading efficiency lower than 1 have been computed using the detailed LAVA code. As a result of international benchmarking, the LAVA code entails uncertainties of about 25%. The safety submission states that spreading will last less than 200s, whatever the case. LAVA code results indicated that 4% of the runs implementing the third case (m_C= 403t; Δt_{inflow} = 178s) may not necessarily meet the design intent because of timescales and final spreading area. These non compliant cases result from bounding scenarios where a viscous corium (due to large initial solid fractions, >20%) flows through a small opening cross section of the melt plug. Assuming that the scenarios with a small opening cross section of the melt plug occur in 10% of the cases, only 0.4% of the potential spreading durations would be greater than 200s.

283 In summary, I consider that the confirmatory analysis using an independent methodology supports the spreading efficiency of the molten material discharge from the transfer channel into the spreading compartment. I do however note that this confirmatory analysis demonstrated a shortfall in some assumptions made in the PCSR methodology. I have therefore raised the following Assessment Finding requesting additional sensitivity assessment to capture the uncertainties associated with the melt plug flow area and the influence of high viscosity on the overall results.

Finding

AF-UKEPR-CSA-20 *– The licensee shall, prior to construction – nuclear island safety-related concrete, provide updated spreading calculations for bounding scenarios employing appropriate viscosities and melt plug opening cross sectional areas*.

4.4 Severe Accident Consequences

4.4.1 Steam Explosion in Accident Conditions

284 A steam explosion is an intense interaction between molten fuel and water in which there is rapid transfer of heat from the molten fuel to the coolant, leading to the heating of water by stable film boiling. Any pressure wave passing through the water then disrupts the vapour film causing direct contact between water and fuel which can trigger a rapid exchange of energy and a phase change which releases energy in the form of a shock wave.

285 The resulting shock wave may cause damage to local structures, or may accelerate a slug of material towards vulnerable structures which may cause damage.

286 Prerequisites for an efficient steam explosion are the absence of large amounts of vapour bubbles (which usually implies significant initial water sub-cooling) or non-condensable gas bubbles, and the presence of the molten material in the form of finely-divided droplets.

287 In the first reviews of the risk of operating nuclear reactors, steam explosions were identified as a significant safety issue in the event of a severe accident, but subsequent programmes of research have been able to demonstrate that the consequences of such events in reactor are likely to be lower than previously thought.

4.4.1.1 EDF and AREVA’s Safety Case

288 A steam explosion may happen, if conditions are favourable, at different stages in the progression of a severe accident, most notably, when melt drains from the core into residual water in the reactor lower head and, following vessel failure, should the melt relocate into a region containing water.

289 The risk of an energetic in-vessel steam explosion is mitigated to some degree by stopping efforts to re-establish safety injection once signs of core degradation are available.

290 Ex-vessel, the reactor pit is intended to be dry at vessel failure, thus precluding a steam explosion in this region.

291 Should these measures fail, EDF and AREVA claim that the likelihood of damage to the containment or the reactor pit structure, from steam explosion, is low based on assessment of possible scenarios for the relocation of molten melt into water pools.

4.4.1.2 Assessment

292 The UK EPR design intent is that water will flow over the molten debris after the debris has relocated to the spreading room. For this scenario, EDF and AREVA cite experiments such as FARO, KROTOS, MACE and OECD-CCI project and physical processes to support their claim that a damaging steam explosion is unlikely. In addition, the cited experiments do not necessarily represent the geometry of the water flow over the molten corium as in the EPR spreading room but configuration of pour of corium into water-filled cavity for which the fuel-coolant energetic interaction could occur with significant fragmentation of corium. However, these experiments collectively represent the key features associated with the phenomena likely to be experienced during the progression of an accident scenario.

293 The perception is that a steam explosion is not a totally incredible event, and so there is a need to assess the damage potential. Thus, I have considered the risk of a steam explosion in both the scenarios of debris relocated in the RPV bottom head (in-vessel) and in the reactor pit and spreading compartment (ex-vessel).

4.4.1.3 In-Vessel Steam Explosion

294 The in-vessel steam explosion has been examined by EDF and AREVA and reported within the safety submissions and the subsequent responses to queries raised during the GDA Step 4 assessment (TQ-EPR-1387, Ref. 9). This subject was jointly reviewed with chemistry topic area from the perspective of the likelihood of a damaging steam explosion.

295 It is widely recognised that the likelihood of a steam explosion, and the associated energy conversion, depends on the properties of the melt. After decades of investigation there is still uncertainty relating to the processes involved. Thus, the state of the art does not permit a major consideration of chemical effects in these processes at this time.

296 There is some doubt whether a steam explosion is possible under the conditions that are likely to prevail in the reactor between water in the lower head and a postulated pour of molten fuel. However, it is prudent to start with the assumption that an energetic melt-water interaction is possible when a melt stream enters water at low pressure; this is the approach taken by EDF and AREVA.

297 EDF and AREVA have considered the loadings on both the upper and lower head following an in-vessel steam explosion. They accept that a steam explosion is possible, but claim that significant damage to the RPV is highly unlikely.

<u>Upper Head Missile Damage</u>

298 The RPV upper head and structures can be at risk from potential missile generation. This could potentially damage the containment in the unlikely event of the upper head failing. EDF and AREVA evaluated the range of energies that could be transferred to a slug of material that would impact the upper head region and compared this with an energy-based criterion for failure of the upper head structures. They concluded that the probability of upper head failure is low because of a combination of circumstances:

- The limited mass of melt that is sufficiently mixed with water to contribute to a steam explosion. The mass is limited because of the calculated relocation rate from the core, the limited depth of water (implying that melt settles out in a short time) and the generation of steam in the mixture region.
- The range of efficiencies for conversion of thermal to mechanical energy.
- Inefficiencies in energy transfer to the slug of material.

299 Their judgments are made on the basis of a mixture of code calculations (e.g. for the melt release rate from the core and the initial mixing with water) and experimental data (e.g. for conversion efficiencies).

300 Overall the analysis presented in the safety submission suggested that there is a very low probability of a steam explosion failing the upper head of the reactor. These probability distributions were arrived at based on expert judgment and are in line with other previous assessments such as that performed for Sizewell B and the SERG-2 review at Ref. 39.

301 The analysis indicates that any likelihood of damage to the RPV upper head region arises from postulated conditions at the upper tails of the probability distributions for melt mass and conversion efficiency, both of which signify conditions that are not very credible. It follows that the likelihood of failure of the upper head is low.

302 This conclusion is consistent with the view of the OECD reported in Ref. 40, which concludes that this mode of vessel failure can be considered resolved from a risk perspective, meaning that this mode of failure is of very low probability and is of little or no significance to the overall risk from a nuclear power plant. I support this view.

Lower Head Damage

303 EDF and AREVA claim that the energy available in the interaction with the water pool in the lower head is significantly influenced through the direction of the melt flow from the core.

304 One possible relocation route of melt will occur from the core into the downcomer through a limited breach of the heavy reflector. The limited breach size and the low gravity head of the in-vessel melt-pool leads to a limited pour rate and hence a limited level of interaction. EDF and AREVA claim that this is the likely scenario.

305 The MAAP code analysis performed by EDF and AREVA supports the heavy reflector relocation pathway. However, an alternative relocation route through the lower core support plate could potentially lead to larger melt pours. I judge that the actual relocation route is uncertain, and I have assessed the consequences of worst case as a precaution.

306 In a recent communication EDF and AREVA have cited the international SERENA calculation exercise on this topic. In SERENA (Ref. 51), the participants attempted to model the initial interaction of melt and water for a diverse range of assumptions, for a relocation flow through the lower core support plate, but with limited flow area. Codes used in SERENA, when applied to the reactor scenario predicted RPV survival. I take this to be an indicative outcome but I have not assessed the validation of these codes. However, I also note that other validation exercises on predicting steam explosion such as CULDESAC have resulted in similar outcomes.

307 I have noted that the SERENA exercise highlights that there are many uncertainties and complexities in modelling corium interaction with water. The major sources of uncertainty

remain the characteristics of the flow regime in the premixing phase, especially void behaviour, and of the fragmentation of corium melts in the explosion phase.

308 To date, the few experiments performed at atmospheric system pressure and subcooled water with small masses of prototypic core melt in the KROTOS facility did not produce an explosion even when an artificial trigger was applied. The few higher pressure experiments performed at system pressures in the range of 0.2 to 0.4 MPa showed very weak explosive events when artificially triggered (Ref. 41). While this is encouraging, there is a need to monitor research in this area and periodically consider its implications for severe accident management.

309 My conclusion is that, EDF and AREVA have presented a case based on current international understanding such that the probability of a steam explosion sufficiently energetic to breech the RPV is very low. This is based on subjective views on melt progression and conversion efficiencies, supported, in part, by limited modelling and the experimental database.

4.4.1.4 Ex-Vessel Steam Explosion

310 Melt can contact water ex-vessel, either in the reactor pit, transfer channel or the spreading compartment. The design intention is that the reactor pit and transfer channel are maintained dry. However in some accident scenarios water may accumulate in the reactor pit. EDF and AREVA have assessed the likelihood that, in such scenarios, there could be local damage to the reactor pit structure. This situation, if it occurs, is conceptually similar to the in-vessel steam explosion, except that now there can be a larger mass and depth of water involved, and the water may be sub-cooled.

311 The UK EPR Severe Accident safety submission considers the ex-vessel steam explosion in the reactor pit basically employing two components: a probabilistic assessment of the energy release, and an analysis of the response of the containment to the shock wave.

In the Reactor Pit

312 A steam explosion immediately following vessel failure is prevented by ensuring that the initial presence of water in the reactor pit is avoided.

313 In response to an enquiry relating to the presence of water within the reactor pit (TQ-EPR-1062 and 1346, Ref. 9), EDF and AREVA have provided information to justify how the reactor pit is kept dry to enable the plant to avoid a steam explosion, recognising that this may occur if the core melt pours into a pool of water. They have stated that water ingression is prevented by mechanical shielding and by the design intent to limit sources of water in connecting compartments. EDF and AREVA further claim that a cavity seal ring is planned to be welded on the RPV flange. This will prevent water ingress from the containment above RPV. However, it may still be possible for water to enter from adjacent compartments alongside the gap around the cooling loops.

314 The accumulation of a substantial depth of water would be required for a high-energy steam explosion. The reactor pit melt plug is not a sealed arrangement and does not seem to be capable of preventing water seeping from the reactor pit into the transfer channel and onward into the spreading area, where there is a water sensor.

315 The operation of the melt spreading compartment water sensor is part of normal operations, and I have not considered this as part of my assessment.

316 I consider that the principles provided in the response are satisfactory in defining the functional water safety provisions for the reactor pit. However, the operational arrangements, sufficient to avoid the presence of a substantial amount of water in the reactor pit will need to be addressed during the licensing process. I have therefore raised the following assessment finding.

Finding

> ***AF-UKEPR-CSA-21*** *- The licensee shall, prior to inactive commissioning – containment pressure test, provide the measure(s) and arrangement(s) for inspection in order to ensure that the reactor pit is kept sufficiently dry*.

<u>In the Spreading Compartment</u>

317 I have considered the possibility that melt-water interactions can occur both when the core melt first enters the spreading compartment and after the compartment is first flooded with water.

318 Like previously in the reactor pit, significant quantities of water in the spreading room must be avoided to limit the risk of steam explosions.

319 In response to TQ questioning (TQ-EPR-1062, Ref. 9), EDF and AREVA have provided information in order to justify how the potential presence of water in the spreading compartment is addressed. They have stated that during normal operations a sensor will be installed in the spreading compartment to detect water and to send an alarm to the Main Control Room in order to shutdown the plant. The safety submission states that dry conditions are not required for the spreading process but the absence of water makes the melt distribution more predictable. I am broadly content with the responses that have been provided in this area.

320 Despite the design intention to maintain the melt spreading area dry prior to melt release from the reactor pit, the safety submission considers the presence of limited quantities of water in this area prior to melt relocation.

321 Based on expert opinion, the probabilistic assessment of the energy release and its quantification requires various assessments and judgements. The information available is based on limited experimental evidence using representative quantities of the materials in the reactor situation. Thus, the phenomena, use of codes and interpretation of insights are broadly similar to those outlined for in-vessel steam explosion.

<u>Re-flooding</u>

322 EDF and AREVA claim that the process of controlled flooding of the spreading compartment is benign.

323 EDF and AREVA cite the MACE experiments (PCSR, Chap. 16.2, Ref. 12) as being appropriate to interactions in the spreading room and suggest that adding water to melt will be benign. However, I note that there are only a limited number of tests in the MACE series and the database is too small to make authoritative judgements so I have obtained independent views. My TSC has considered relevant events and concurs that melt-water interactions in the spreading room would be generally benign. I judge that general experience suggests that the likely outcome is stable film boiling on the crust of the debris.

324 The withstand capabilities of the spreading compartment and other aspects of the containment in close proximity to a local substantial pressure loading generally fall within other disciplines.

325 EDF and AREVA have presented the risks and consequences of the steam explosion and examined the appropriate research findings.

326 I acknowledge that the probability of core damage and subsequent damage to the containment by a steam explosion in the event of a severe accident is lower than previous generations of PWRs.

327 In summary, it is not unreasonable to conclude that the likelihood of an ex-vessel steam explosion, sufficient to cause major disruptive melt dispersion, is low.

4.4.2 Corium Re-criticality

328 One of the essential safety functions that needs to be addressed is the ability to shut down the chain reaction and retain the core subcritical.

329 The potential for re-criticality is one of the hazards to be considered when the core configuration is lost. This requires consideration of the pool of molten debris formed once the core has relocated to the RPV lower head and the corium melt as it moves from the RPV into ex-vessel positions.

330 The PCSR does not comprehensively address the potential for re-critically of the relocated molten material to occur at any location including the core and within the CMSS. However, in a response to RO-UKEPR-44.A3 EDF and AREVA have presented analysis of the margin to re-criticality calculated using CASMO 4, Ref. 42. I have not chosen to sample the validation of the code for the purpose. I base this on the fact that this is a commercial code, widely used for criticality assessment, and a large body of validation evidence exists.

331 However, I note that the analysis uses a conservative value for the mean fuel enrichment in a core at the start of a cycle of irradiation and makes conservative assumptions about the separation of absorber material.

332 The composition of the melt is analysed using MAAP and COSACO codes and the assessment of this topic is reported in the Chemistry GDA Step 4 assessment report Ref. 26.

333 In addition, EDF and AREVA in response to an enquiry relating to potential re-criticality ex-vessel, (TQ-EPR-1067, Ref. 9) provided additional explanation that the risk of the corium returning to a critical configuration is sufficiently low provided that the reactor pit and the spreading compartment is dry or any water contains boron at the concentration designed for the emergency cooling systems. This methodology makes use of conservative assumptions for porosity, fragmentation and temperature within the melt pool.

334 In my assessment of the PCSR, the review of the OSSA was excluded, the assessment of which will be performed during the site licensing activities.

335 I am satisfied that provided the corium does not come into contact with non borated water, that it will remain subcritical. However, I consider that the risk of re-criticality due to the relocated molten material and its progression within the CMSS should receive further examination by the future licensees.

Finding

> ***AF-UKEPR-CSA-22** - The licensee shall, prior to construction – nuclear island safety-related concrete, provide a comprehensive examination of re-criticality for all reasonably foreseeable conditions during the transient progression and within the CMSS.*

4.5 Containment Combustible Gas Control System

336 During a number of design basis and potential severe accident sequences, the possibility exists for the generation of hydrogen-rich atmospheres within the containment of any Light Water Reactor (LWR). The major concerns regarding hydrogen are that the pressure or thermal loads from combustion may damage containment or that important safety-related equipment may be damaged. In order to assess the possible threats, it is necessary to understand how hydrogen is produced, how it is transported and mixed within containment, and how it combusts.

337 The potential for hydrogen build up under design basis or severe accident conditions comes principally from the possibility of fuel cladding oxidation. In the context of the containment, the concern is that hydrogen produced is released from the primary circuit at some point in the transient into the containment building where combustion could result in loss of the containment function.

338 Success criteria applied to design basis faults mean that global conflagration can be discounted and the effects of local releases from the primary circuit are the principal concern.

339 The need to manage hydrogen stems mainly from severe accidents where hydrogen can be released in substantial quantities as the fuel overheats. My assessment has focused on this aspect and has considered whether reasonably practical measures have been taken to mitigate the risk and whether the analysis employed has been suitably substantiated.

340 The UK EPR containment is designed without active measures to mix gasses within containment and relies on natural circulation to achieve this. Furthermore, the UK EPR design features the CONVECT system, which allows the transfer of the two room containment (‘accessible’ and ‘inaccessible’ parts during normal operations) into a single containment volume. This separation is convenient for plant operations, but complicates the combustible gas management during an accident by delaying dilution and mixing.

341 In certain accident sequences, gas is released into the smaller inaccessible parts of the containment, initially resulting in high concentrations in these areas. To counter this effect, the operation of the CONVECT system that includes the opening at the top of the SG compartment is expected to promote commencement of a global natural convection within the enlarged containment. I acknowledge that the natural convection within the containment under accident conditions is aided by the operation of a series of mixing dampers and rupture panels. For this system to work effectively, efficient mixing is required for the content of the smaller inaccessible volume to be diluted in the much larger accessible part.

342 I have examined the design of the containment rooms to satisfy myself that they meet the key principles in the SAPs and have examined the analysis methods to satisfy myself that they meet the requirements of SAPs FA.15, FA.16 and FA.18 requiring that they adequately represent the important processes taking place and that they have been suitably validated.

4.5.1 EDF and AREVA's Safety Case

343 The strategy for containment hydrogen mitigation and management within the UK EPR containment is explained in the PCSR (Ref. 43, Section 6.2.4) and the corresponding SDM (Ref. 53)..

344 Chapter 16 of the PCSR identifies that in design basis accidents, a fault study acceptance criterion limits the level of tolerable cladding oxidation to 1% of the amount generated if all the active part of the cladding were to react. This limits the global hydrogen level to below 4% by volume and prevents the possibility of global combustion in containment.

345 The design objective of the UK EPR containment is to ensure that a global hydrogen level of 4% by volume under dry conditions is avoided. This is principally achieved by the large free volume of the containment.

346 The local concentration may exceed this level and flammable levels of hydrogen can not be excluded, but mixing is promoted by the provision of the CONVECT system that includes carefully positioned foils and dampers which are designed to fail/open in the event of a loss of coolant and to promote circulation within the containment building as a whole. This system is designed to ensure that the extent of high hydrogen concentrations is limited and that any combustion will not lead to damage to the containment shell. Here the goal is to avoid reaching a hydrogen concentration of 10% by volume to minimise the risk of global detonation. In addition, analysis is required to demonstrate that local flame fronts do not accelerate unacceptably. The intent is, to comply with the criterion for Deflagration to Detonation Transition (DDT) by evaluating the combustion process, in particular with regard to pressure histories and flame velocity.

347 In severe accidents, the level of hydrogen within containment is limited by passive autocatalytic recombiners which reduce global levels of hydrogen to acceptable levels. These are placed principally in equipment rooms, but also under the dome to cope with the possibility of stratification. Analysis to justify installation of the PARs is found in Ref. 44.

348 The approach to justifying the hydrogen mitigation measures has been to examine scenarios of two categories, "representative scenarios" to demonstrate the efficiency of the system, and "bounding scenarios" - selected to demonstrate the robustness of the design concept.

349 The representative scenarios were analysed with certain pessimistic assumptions to demonstrate the leak tightness of the containment, while the bounding scenarios are employed to demonstrate the robustness of the severe accident control and mitigation measures and to show that no "cliff edge" effects exist. The bounding scenarios were analysed employing realistic assumptions.

350 A common assumption for representative scenarios is the complete failure of the safety injection whilst for bounding scenarios late injection is considered.

351 Analysis of the hydrogen risk examines the potential energy available on the basis of AICC and the potential of flame acceleration. Analysis of DDT is based on the results of calculations using the GASFLOW code.

352 Flame acceleration and the risk of DDT are assessed by applying experimentally proven criteria. These criteria provide the link from analysis to the experimentally based knowledge. Several criteria treat the non-occurrence of flame acceleration and thus of

fast deflagration. These criteria are used to eliminate fault sequences from further consideration.

353 If fast combustion can not be excluded by the criteria, explicit calculation of the combustion process and the resulting dynamic mechanical loads on containment structure is made using COM3D.

354 The analytical approach adopted for severe accidents uses several steps and a series of codes:

- MAAP4 provides mass and energy release from the primary circuit (including molten corium at vessel failure).
- COCOSYS determines pressure and temperature in the containment and global aspects of hydrogen mitigation.
- GASFLOW evaluates gas distribution and hydrogen combustion and allows the assessment of potential flame acceleration.
- COM3D is used to predict the combustion process in more detail and allows the assessment of potential flame acceleration and dynamic pressure loads.

355 Each code has been validated by a series of separate effects and where appropriate, integral testing. These are further discussed at Section 4.7.

4.5.2 Assessment

356 The chemical aspects of hydrogen generation and combustion are addressed in Ref. 26. This deals extensively with the design and operation of the PARs and therefore I have chosen only to examine this aspect briefly (see Section 4.6).

357 Hydrogen-air-steam mixtures can burn in several ways dependant upon the conditions; namely as diffusion flames, slow deflagrations, accelerated flames and detonations. EDF and AREVA consider each of these combustion modes for UK EPR.

358 Diffusion flames (or stationary flames) result when combustible gases are released into an oxygen-rich environment; creating a flammable plume. In UK EPR, these are predicted to occur during molten core concrete interaction when released gases are above the auto-ignition temperature in the reactor pit and spreading area. This is the least damaging combustion, provided that the thermal loads do not result in harm to surrounding equipment.

359 Deflagrations and detonations are rapid burning of often pre-mixed gases. The speed of the combustion is important in determining the consequences. Detonations are potentially the most damaging because much of the energy can be present in the form of a shock wave. Thus, it is important that the transition from deflagration to detonation is avoided.

360 Accelerated (or fast deflagration) flames can be considered intermediate between deflagration and detonation and can in themselves provide relatively high loads.

361 The standard assumption is that if the conditions for a deflagration exist, then this will be triggered at the most adverse time. EDF and AREVA have attempted to follow this practice in their analysis.

4.5.2.1 Reported Analysis

362 The approach to justifying the hydrogen mitigation measures has been to examine scenarios of two categories:

- Representative scenarios - selected mainly for their likelihood of occurrence - analysed conservatively, and
- Bounding scenarios - selected for phenomena that might occur to aggravate the hydrogen risk, such as re-flood at a most unfavourable moment, or delayed depressurisation - used to demonstrate the robustness of the concept.

363 The representative scenarios are generally SBLOCA at different locations. As containment failure risk results mainly from fast deflagration and DDT, EDF and AREVA focus on scenarios with fast secondary side cool-down, leading to low steam concentration in the containment, likely to favour flame acceleration.

364 The bounding scenarios are characterised by delayed depressurisation or active re-flood where the hot core is flooded (by accumulators or by the Safety Injection System) resulting in the release of a large amount of hydrogen, generated at a high rate. Break size and time of delay have been selected, based on parametric studies, to maximise hydrogen production.

365 This approach appears reasonable as a way to provide a demonstration of system performance.

4.5.2.2 Representative Sequences

366 In the representative sequences, the predicted consequences have been shown to be tolerable with relatively straightforward analysis.

367 MAAP, COCOSYS and GASFLOW were used to calculate the pressure in the containment assuming Adiabatic Isochoric Complete Combustion (AICC). This approach provides an increment to the calculated containment pressure without combustion. The practice is penalising from a temporal and thermal perspective because it assumes combustion of all available gas with no heat losses to structures.

368 EDF and AREVA claim that the AICC pressure will remain below the containment design pressure for “representative scenarios”. However, for the “bounding scenarios” the Adiabatic Isochoric Complete Combustion pressure may exceed the containment design pressure for a relatively short period. The supporting analysis aims to demonstrate that at least the AICC pressure remains below the containment test pressure, being 1 bar above the design pressure.

4.5.2.3 Bounding Sequences

369 Three “bounding scenarios” were investigated:

- Loss of off-site power with failure of all Diesels and reflood at the most unfavourable moment to analyse thermal loads.
- SBLOCA with delayed depressurisation to investigate dynamic- and thermal loads from accidental hydrogen combustion, also at the most unfavourable moment.
- SBLOCA with re-flood at the most unfavourable moment to analyse potential combustion.

370 The scenarios selected met my expectations and I consider them suitable for examining the effectiveness of the hydrogen mitigation measures.

371 Large break LOCA is not limiting for the hydrogen risk; however, the overall efficiency of the recombiner was analysed with COCOSYS.

372 Generally the recombiners were not explicitly modelled in the analysis, but this has been justified by more detailed analysis of selected sequences.

4.5.2.4 Results of Pressure Loads

373 For bounding scenarios AICC pressure is slightly higher than the design level in some sequences considered. However, pressure calculated with GASFLOW considering the temporal and spatial distribution, is significantly lower because at the level of hydrogen concentration predicted in some regions, complete combustion will not occur Ref. 45.

374 In the bounding sequences, the combustion criteria indicated the possibility of local fast deflagration and even DDT in the period shortly after onset of hydrogen release. Further analysis was required using the COM3D code. This demonstrated that damagingly high local pressure peaks did not occur on the containment shell. This is because flame acceleration within the SG compartment is limited: radial venting dissipates the flame and deceleration of the flame front occurs in the dome.

375 The DDT risk is limited to the early period after onset of hydrogen release, where the distribution of combustion gasses is inhomogeneous and regions of high concentration exist in the equipment rooms. At this time little hydrogen is in the dome.

376 Later, when the concentration in the dome rises, the overall gas distribution is quite homogeneous with hydrogen concentration well below 10 vol%. This reduction in concentration is mainly due to gross convection but also the recombiners provide some mitigation.

377 Although detailed analysis indicates no threat to the containment dome, high dynamic pressure differences across internal walls can result from fast hydrogen combustion in the equipment rooms.

378 In the initial design analysis, the flame acceleration was particularly pronounced for the scenario with low steam concentration in the upper pump room of the affected loop, because this room was a dead space. These calculations led to the provision of additional openings between the upper pump room compartments and the adjacent SG compartments.

379 EDF and AREVA remain dependent on detailed computational modelling to justify the acceptability of loads on containment structures in the event of hydrogen burn. Furthermore, the design of the equipment rooms remains an area which requires detailed assessment.

380 Generally in single containment designs, rapid depressurisation may release large quantities of hydrogen at a point in containment, but the associated turbulence helps to limit the likelihood of flammable mixtures reaching concentrations for extended local flame acceleration. The adequacy of the mixing arrangements for UK EPR therefore needs to be demonstrated by a robust analysis.

381 I did not find the documentation initially provided by EDF and AREVA sufficient for my assessment. Consequentially I issued Regulatory Observation 78 requiring EDF and AREVA to provide sufficient analysis of the behaviour of the UK EPR containment, during

accident conditions which involve combustible gas releases, to provide assurance that the design proposed is acceptable.

382 The response to this RO arrived outside the planned time interval for Generic Design Assessment and will require further detailed assessment. I have examined it briefly and am generally satisfied. However I anticipate that further information will be required to demonstrate that nothing further can reasonably be done and therefore I have raised an assessment finding requiring that the licensee provide further justification and clarification of the safety case. I would expect to see some optioneering considering whether procedure or plant changes could further mitigate the risk.

383 EDF and AREVA state that the effects of ignition of hydrogen in the confined equipment rooms, can potentially develop into a fast deflagration but the time interval for which this possibility exists is small. The resulting dynamic pressure loads on internal wall and structures, as well as on the containment shell, are discussed in Ref. 46.

384 Analysis of the thermal loads on containment structures is presented in Ref. 47. EDF and AREVA argue these to be of minor importance, because the combustion is primarily located in the equipment rooms. Once the flame front propagates into the containment dome and the upper annular rooms, it dissipates and the associated temperature loads are acceptable.

385 Without combustion, the temperature loads on containment walls and structures are only moderate. Global convection inside the entire containment atmosphere effectively distributes the associated temperature loads from hot steam over large areas, including the operating rooms.

386 The temperature loads from the recombination of hydrogen are more localised. Each operating recombiner emits a hot exhaust plume which rises due to buoyancy and chimney effects. A detailed assessment of the exhaust gas temperatures and the quantification of surface temperatures at the surface of the containment shell liner in areas close to the recombiners demonstrate that the distance between recombiner exhausts and the liner is sufficient to avoid critical temperature loads.

387 The hot exhaust plumes can potentially impinge on the steel liner that covers the interior of the containment shell. As the liner has a low thickness, so it can potentially deform plastically “known as blistering” if it is very strongly heated locally. Furthermore, in its cylindrical part, there are penetrations for cables, ducts and pipes to the containment exterior which are also sensitive to excessive temperature loads. The positioning of Recombiners in the UK EPR containment addresses the requirement to keep a sufficient distance between a recombiner exhaust and critical structures.

388 In my assessment of the UK EPR tolerability to risk, I have also considered the results of independent confirmatory calculations performed by other international regulators representing a tolerable risk to the containment. Considerations of the structural impact of these loads have been examined as part of the Civil Engineering Assessment of the UK EPR design, reported in Ref. 28.

389 In summary, since much of the material presented by EDF and AREVA has arrived too late for detailed assessment within GDA, I feel that further examination of the material is necessary during the site licensing phase. In particular, I see the need to consider whether it is ALARP to take additional measures to limit peak hydrogen concentrations. It seems that depressurisation of the primary circuit through relief valves presents a foreseeable scenario for release of hydrogen to containment and I would expect that this should be optimised to minimise the associated risk.

Finding

AF-UKEPR-CSA-23 *- The licensee shall, prior to construction, nuclear island safety-related concrete, justify that the measures taken to mitigate hydrogen-related risk set out in response to RO-UKEPR-78 are ALARP and, in particular, that there are no reasonably practical measures that would increase mixing of hydrogen plumes during a delayed depressurisation of the primary circuit in a severe accident*.

4.6 Containment Demands Following Severe Accident

390 The containment has been designed to provide a reliable and leak-tight barrier following the on-set of core damage. The containment is further designed to be a secure barrier to the release of fission products. The containment heat removal system is present to control the pressure and thermal (temperature) demands. The UK EPR containment is double walled sitting on basemat raft foundations. Thus, it is a robust structure. Leakage is strictly controlled to ensure minimal leakage levels. The volume is ~80,000m^3. The internal volume of the Containment Building is larger relative to most existing PWR plant designs which is a positive beneficial effect in terms of pressure and temperature demands relative to power and decay heat levels. Although there is a grace period for 12 hours for no active containment pressure control such as the CHRS, for most fault scenarios, the containment design pressure will not be reached within 24 hours providing additional time to allow off-site emergency response.

391 The containment heat removal assists in the control of pressure and temperatures within the containment and the hydrogen mitigation system of 47 autocatalytic recombiners controls the build-up of hydrogen. The two volume into one volume concept promotes containment mixing vital to the effective control of local ‘pockets’ of hydrogen. Hydrogen is generated from known phases of the accident progression.

392 In terms of the hydrogen behaviour it is lighter than air. Hydrogen has a rapid diffusivity which is 3.8 times faster than natural gas which means that when released, it dilutes quickly into a non-flammable concentration. Hydrogen rises two times faster than helium and six times faster than natural gas at a high speed. Thus, the laws of physics prevent hydrogen from lingering near a leak or maybe within a fairly closed sub-compartment with minimum opening (Vents, dampers, foils etc).

393 The whole area of the phenomena within containment may be considered for further study in the next phase given the lateness of the confirmatory analyses to ensure that all systems operate in the manner intended within the various code models. Best Practice will dictate very detailed modelling from local and more global perspectives.

394 The issue of containment hydrogen distribution is very important in a design with many compartments. Hydrogen is very light and may diffuse through heavier gaseous substances. In spaces without inherent convection currents, hydrogen may stratify. If hydrogen consolidates in high concentrations then a combustion risk will occur. EDF and AREVA claim detailed comprehensive modelling using the CFD code that will justify the hydrogen claims will be available in Q1 2011. The output will require detailed study to examine for localised ‘hot’ spots of hydrogen.

395 An important distinction that needs to be made here is that EDF and AREVA provided specific analysis for UK EPR in April 2011. This was the subject of RO-UKEPR-78.A2, for which the response was late to be included in GDA Step 4 assessment report and will

now be the subject of a related GDA Issue (**GI-UKEPR-RC-01**) raised in the Reactor Chemistry Assessment report (Ref. 27).

396 However, EDF and AREVA did provide detailed information on the intended approach and background for the analysis which will support the design in addition to some modelling results from the design phase of the generic EPR which are very similar to the UK EPR. This formed the basis of the assessment that follows. Irrespective of this, the final UK EPR analysis needs to be reviewed to ensure consistency with the conclusions of my assessment. This topic is further covered within the Reactor Chemistry assessment report and has resulted in a related Assessment Finding.

4.6.1 Impact of Fission Products on the Performance of the Passive Auto-Catalytic Re-combiners (PAR)

397 Depletion of the coolant leading to uncovery of the reactor fuel and its degradation in a severe accident can result in relocation of molten material into the reactor pit leading to the erosion of concrete within the cavity wall which produces sustainable chemical heating. The degradation of fuel and the MCCI will result in production and release of significant quantities of non-condensable gases such as hydrogen and some CO and CO_2 into the containment environment. The quantities of hydrogen generated and released into the containment has been predicted and presented in the safety submissions. The UK EPR safety submission also recognises the presence of CO and CO_2 and assumes this will not adversely influence the performance of the PARs.

398 The PARs are strategically distributed within the containment to reduce the combustible gases and will be exposed to all the potential contaminants in the atmosphere. Core degradation will also result in fission product release and recent research has shown that metal iodides in aerosol, principally CsI, entrained into the PARs can result in conversion of Iodine aerosol to gaseous iodine inside a PAR due to temperature decomposition (Sabroux, et al. Ref. 48).

399 PAR conversion of airborne CsI to I_2 and regulatory/safety limits on gaseous iodine levels may pose safety concerns with regard to control room dose and environmental releases. This might also have implications on equipment qualification with respect to dose rate. Ongoing research in this area should be reviewed and the findings of this research considered with respect to implications on resultant doses due to normal containment leakage (ie. Control room and site boundary) and on dose to critical equipment.

4.6.1.1 Assessment

400 The concern relating to the performance of the PARs and generation of gaseous Iodine was raised with EDF and AREVA at (RO78 A1 – Ref. 10). In their response EDF and AREVA have informed HSE ND of their intention to continue research on all aspects of the PARs performance in the environment likely to be experienced during accident conditions.

401 The adequacy of the PAR design has been assessed by the chemistry topic area review which will also include an examination of the residual iodine risk associated with the acceptability of this equipment. The result of this will be reflected in the relevant topical area report at Ref. 26.

402 The operability of CHRS is integral to the successful control of temperature and pressure within the containment, and will also assist in mixing of hydrogen throughout the containment in the long term. The research on hydrogen mitigation devices seems to be

comprehensive and many issues that were raised always seem to have been addressed. I am satisfied the issue of flow distribution within containment and the containment performance aspects will be addressed comprehensively when the very detailed CFD modelling is available.

403 In summary, the current knowledge suggests that the proposed PARs are capable of performing the expected function within the containment environment, although additional confirmatory experimental work is required to provide greater assurance that fission product poisoning of PARs is unlikely to adversely influence the operational capabilities of the PARs and to demonstrate the adequacy of the design over the full operating range.

Finding

AF-UKEPR-CSA-24 *- There are a number of observations made with regards to the operational requirements for PARs during accident scenarios. Given the significance of the equipment to hydrogen concentration management during accident progression, the licensee shall, prior to construction - nuclear island safety-related concrete, provide additional justification that;*

- *considers the poisoning of the PARs by the released fission products informed by the outcomes of the planned experimental programme, and*
- *demonstrates the continued operability of the PARs in prolonged accident scenarios.*

4.6.2 Control of Radiological Releases

404 The UK EPR includes design measures aimed at controlling the release of radioactivity resulting from design basis and severe accidents. These are primary depressurisation in the event of an accident to prevent containment bypass accidents via steam generator tube rupture (SGTR) and through the control measures for leakages out of containment to the outside environment.

405 Discharges are closely coupled to internal containment pressure, annulus pressure, performance of mechanical equipment, performance of filters and containment penetration seals. The containment volume and the heat removal system with its diverse options are also important in managing containment environment pressure increases during prolonged accident scenarios.

406 The containment design includes a double shell with an inter-space such that in the event of an accident with radiological releases there are two primary routes from the containment:

- Leakage into the inter-space where there is a negative pressure relative to the outside environment with the concept known as Annulus Ventilation System (AVS). The maximum total containment leakage rate from inner containment at design pressure and temperature is in the order of 0.3% vol/day.
- Leakages which are not collected in the annulus space enter the peripheral buildings and are filtered before being released. If this occurs, there are various concerns associated with the subsequent access, contamination, equipment qualification, clean-up etc. The discharges are eventually filtered through particulate air and high efficiency Iodine filters.

4.6.2.1 Assessment

407 The pressure and temperature conditions inside the containment have a key role in the performance behaviour of the AVS. The operability of the containment heat removal system using the spray in response to a severe accident should help to ensure the integrity of the containment. Thus, the AVS cannot be examined in isolation of many other factors such as the operator actions, operability of containment mitigation systems such as PARs and Sprays, integrity of penetration, vacuum pumps, failure to initiate sprays and assumptions on penetration seal effectiveness/deterioration through plant life.

408 In addition, the containment external conditions are subject to local winds and should be representative of UK site conditions. This will influence the performance of the AVS and challenge the maintainability of the negative pressure in the inter-space, radiological releases profiles, containment external pressure, plume speed and direction.

409 Containment extreme conditions will also impact the short, medium and long term operability of the containment in terms of internal containment pressure and temperature demands. The requirements are to remain within the design limits of 5.5 bar and 170°C for both internal fault conditions and external hazards. The analysis of the environmental conditions within the annulus and peripheral buildings would therefore need to be provided for UK site specific conditions.

410 In my assessment, I have recognised the importance of the CHRS and the spray system that is required to control the long term containment pressure and temperature within the containment. I have also noted that UK EPR design does not offer a filtered discharge facility to vent the containment and therefore provide a passive and diverse method of pressure control.

411 In technical exchanges, EDF and AREVA indicated that the EOPs recommend discharging into the adjacent buildings as an alternative to a filtered discharge.

412 Although no additional information is provided to justify this alternative venting route, I consider that this strategy could lead to increased radiological releases following a severe accident to the peripheral buildings, limiting access for recovery and potential use of equipment.

413 In TQ-EPR-1385-01 (Ref. 9), I requested EDF and AREVA to outline the measures that are proposed for this alternative venting strategy during a severe accident. I also requested clarification of the proposed operational philosophy and whether the risk from the proposed concept is ALARP.

414 The response states that EDF and AREVA are currently reviewing the feasibility of alternative containment venting by a containment penetration in accident scenarios as a means of reducing the containment pressure in the long term.

415 Overall, I expect that the EPR project should identify a design which reduces risks in this area as far as reasonably practicable. I am therefore raising an Assessment Finding requesting that a potential licensee demonstrate why the proposed design is ALARP.

Finding

> ***AF-UKEPR-CSA-25*** *– The licensee shall, prior to construction – nuclear island safety-related concrete, provide the available measures to limit the containment pressure, in the event of a severe accident leading to the failure of the CHRS, to prevent uncontrolled radiological releases from the primary containment*.

4.7 Codes and Methodologies

4.7.1 Hydrogen Analysis Codes

416 The demonstration of the effectiveness of the hydrogen risk mitigation measures is based upon a sequential calculation process as described in Ref. 47. These codes build in complexity and detail throughout the sequence. The main steps, along with the codes used, are;

- Initial screening analysis of a large number of calculations of the in-vessel phase with the Modular Accident Analysis Programme (MAAP4) code. This produces the mass and release rate of hydrogen-steam mixtures.
- Analysis of the ex-vessel phase of the accident using COSACO where required. This is the equivalent of the MAAP4 analysis for the ex-vessel phase of a severe accident and produces the generation and release rates for hydrogen and MCCI gases.
- Analysis of the containment performance using COCOSYS, taking the mass and energy release values derived from MAAP4 and COSACO. This stage provides more detailed information on the hydrogen depletion rate and average gas concentrations in the containment.
- Analysis of a further subset of key scenarios with the Computational Fluid Dynamics (CFD) code GASFLOW. This provides more detailed analysis of, for example, the containment atmosphere mixing process, PAR performance, and thermal loads from combustion and recombination on structures.
- Where the results of GASFLOW analysis indicate that the conditions are likely to exceed the criteria for flame acceleration, the same key scenarios are analysed with the CFD code COM3D. This is used to demonstrate that Deflagration to Detonation Transition does not occur in the particular geometry and that the combustion pressure loads do not compromise the containment or inner structures important to safety.

417 This analysis procedure is illustrated below, Ref. 47.

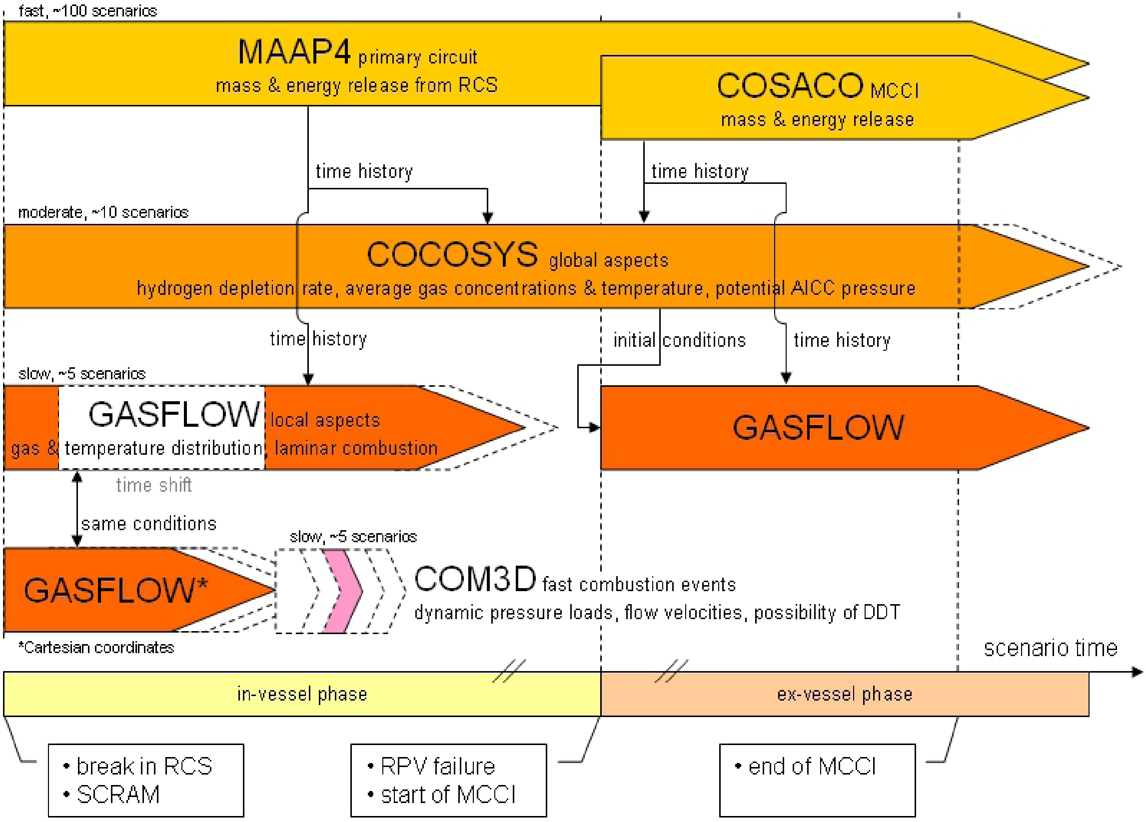

Figure 4: Road Map of the Codes Utilised in Support of the Severe Accident Analysis

418 The MAAP code is used for severe accidents generally and is discussed in Section 4.3. The generation of hydrogen is dependent on the temperature of the cladding and the flow rate of steam and therefore requires similar models to those used in MAAP for other purposes.

419 COCOSYS is only used for a limited number of sequences and given the uncertainty associated with ex-vessel scenarios; I have chosen not to sample these sequences.

4.7.1.1 GASFLOW

420 GASFLOW is a finite-volume computational fluid dynamics code developed at Los Alamos National Laboratory in the USA and Forschungszentrum Karlsruhe (FZK) in Germany for predicting the transport of a gas atmosphere consisting of various gas species as well as the recombination and combustion of hydrogen. The code is validated against a variety of experiments, and the results of these validation efforts are summarized in an assessment manual.

421 The code models the flow of gas and steam in an arbitrary set of rooms, with heat transfer by convection conduction and radiation. If required, the combustion model is initiated at a user-defined time and the development of the flame is predicted. The combustion rate is evaluated using an Arrhenius law.

422 GASFLOW has the ability to model the performance of recombiners.

423 References provided in the documents supplied suggest a complete set of documents meeting our general requirements (Ref. 49), but these documents were not supplied in the response to regulatory observation and therefore our assessment of the code is not complete. I have therefore issued an assessment finding requiring that suitable documentation be provided.

424 I have examined some documentation published by the code authors in the public literature and these give me confidence. In particular, a joint research project was carried out in the EU 4th Framework Programme with the goal to develop verified and commonly agreed physical and numerical models for the analysis of hydrogen distribution, turbulent combustion and mitigation. The conclusions of this were generally positive. However, they found that:

425 "Limitations of the present combustion models and need for further validation do not allow fully quantitative predictions of the detailed containment loads under all conditions. However, they allow studies of the complex turbulence/chemistry interaction processes taking place in realistic large-scale 3D geometry configurations", Ref. 50.

426 This assessment covered both GASFLOW and COM3D and supports my judgement that a suitable case can be made.

4.7.1.2 COM3D

427 COM3D is a special-purpose code used for flame modelling. It predicts the time-dependent development of a shock wave in a combustible gas mixture. The documentation supplied to date to substantiate the use of the code is not considered to be sufficient and needs to be supplemented. I am aware that the code has a detailed combustion model that it has been calibrated against experimental data on flame velocity.

Finding

> ***AF-UKEPR-CSA-26*** *- The licensee shall, prior to construction – nuclear island safety-related concrete, provide a comprehensive set of documentation for the GASFLOW and the COM3D codes used in support of the PCSR. This should include, but not be restricted to:*
>
> - *Detailing the modelling used,*
> - *Guidance on the code limits of applicability, its use and qualified uncertainty allowances, and*
> - *Substantiation of the codes' validity by comparison against measurements and independent analysis.*

4.8 Classification and Categorisation

Assessment

428 The UK EPR plant contains instrumentation systems that are utilised to inform the operators to take the appropriate actions to enhance the safety of the plant during various accident scenarios. The equipment employed would need to have an appropriately assigned safety classification.

429 EDF and AREVA identify four types of safety functions in the November 2009 PCSR; F1A, F1B, F2 and non-classified. An F1A safety function is a function that is required for a PCC event to reach the controlled state. An F1B safety function is a function that is required to reach the safe shutdown state. F2 safety functions are claimed for RRC-A and RRC-B sequences. A system is classified F1A, F1B, F2 or non-classified according to the classification of the highest integrity safety function it must perform. Therefore a system delivering a safety function required for severe accidents had a minimum classification of F2.

430 During Step 4, in response to the requirements of RO-UKEPR-41 and RO-UKEPR-43, EDF and AREVA have undertaken to migrate over to an alternative categorisation and classification system, consistent with that used in the UK. This is the subject of Fault Studies GDA Issue **GI-UKEPR-FS-02** and Cross-cutting GDA Issue **GI-UKEPR-CC-01** (see Refs 32 and 52 respectively). The practical effect of this new classification is that System, Structure and Component (SSC) identified in the Fault Studies chapters of the November 2009 PCSR are effectively mapped over from F1A, F1B, and F2 to Class 1, Class 2, and Class 3 respectively.

431 The emergency operating procedures change to severe accident management on detection of the Core Outlet Temperatures (TCOT) at 650°C, at which time, if not already done so, the operator is expected to initiate the PDS to reduce the RCS pressure under controlled conditions.

432 The thermocouples at core outlet have operational functions associated with monitoring core conditions informing the operators. The PDS is used for reducing primary circuit pressure and discharging coolant during "feed and bleed" for design basis accident. Both these systems are therefore designed to be used during design basis accident conditions prior to a severe accident developing.

433 The PDS is claimed in the design basis safety case to provide a diverse means of protection against some PCC faults. HSE ND's expectation of the safety classification for a SSC that makes a significant contribution (but not the principal means) to fulfilling a Category A safety function is Class 2. As a result, EDF and AREVA are currently assessing the adequacy of the Class 3 classification through the cross-cutting GDA Issue **GI-UKEPR-CC-01** (see Ref. 52).

434 In summary, the thermocouples measuring the core outlet temperatures are to inform the operator to initiate the PDS that is intended to reduce the primary system pressure in a controlled manner during design basis accident scenarios and in the unlikely event of these developing into a severe accident. The safety classification of the instrumentation for the management of DBA should be appropriately assigned to this more limiting case. EDF and AREVA are expected to address the issue of classification through Cross-cutting GDA Issue **GI-UKEPR-CC-01** (see Ref. 52).

4.9 Overseas Regulatory Interface

435 In accordance with this strategy, HSE collaborates with overseas regulators, both bilaterally and multi-nationally.

436 HSE's Nuclear Directorate (ND) has formal information exchange arrangements to facilitate greater international co-operation with the nuclear safety regulators in a number of key countries with civil nuclear power programmes. These include:

- the US Nuclear Regulatory Commission (NRC),

- the French Nuclear Safety Authority - L'Autorité de sûreté nucléaire (ASN), and
- the Finnish Nuclear Safety Regulator (STUK).

437 HSE ND also collaborate through the work of the International Atomic Energy Agency and the OECD Nuclear Energy Agency (OECD-NEA). ND represent the UK in the Multinational Design Evaluation Programme (MDEP) - a multinational initiative taken by national safety authorities to develop innovative approaches to leverage the resources and knowledge of the national regulatory authorities tasked with the review of new reactor power plant designs. This helps to promote consistent nuclear safety assessment standards among different countries.

438 Interface with other international regulators has been principally by bilateral contact which has helped me to share the latest developments in my topic area and assign priorities to technical issues. The contacts were enabled through OECD Nuclear Energy Agency working group meetings in the context of the Multinational Design Evaluation Programme (MDEP).

439 Regarding the sump filters performance issues; the validation work by the US NRC has informed the regulatory decision making. The recent sump baskets and filters qualification testing in support of the EPR design by AREVA have been observed by the US NRC, results of which has been discussed at the MDEP meetings. It should be noted that the US NRC has taken a leading role in establishing a consensus in this topic area.

440 Discussions with STUK have provided useful information and the results of its independent confirmatory analysis across a wide range of issues such as: containment hydrogen mixing and core stabilisation test data has reduced the need for extended scope of confirmatory analysis.

441 The formal contact has been supplemented by attending Joint OECD/NEA – EC/SARNET Workshops. Such meetings and workshop have provided useful background information for judgements.

442 ND is also a member of the Code And Maintenance Programme (CAMP) and the Cooperative Severe Accident Research Programme (CSARP) which are aimed at sharing and supporting US NRC code development activities. ND has also funded the Health and Safety Laboratory (HSL) to perform CFD benchmark activities as part of the OECD international standard problem ISP 49 on the hydrogen distribution in containment following a severe accident.

5 CONCLUSIONS

443 This report presents the findings of the Step 4 Fault Studies - Containment Thermal Hydraulics Response and Severe Accidents assessment of the EDF and AREVA UK EPR reactor.

444 To conclude, I am broadly satisfied with the claims, arguments and evidence laid down within the PCSR (Ref. 14) and supporting documentation for the Fault Studies - Containment Thermal Hydraulics Response and Severe Accident which is listed in the SML (Ref. 15). I consider that from a Fault Studies - Containment Thermal Performance and Severe Accident view point, the EDF and AREVA UK EPR design is suitable for construction in the UK. However, this conclusion is subject to assessment of additional information that becomes available as the GDA Design Reference is supplemented with additional details on a site-by-site basis.

5.1 Key Findings from the Step 4 Assessment

445 In my assessment of the containment hydraulics performance and severe accident of the UK EPR, I have a raised 26 Assessment Findings that need to be resolved, as appropriate.

5.2 Assessment Findings

446 I conclude that the Assessment Findings listed in Annex 1 should be programmed during the forward programme of this reactor as normal regulatory business.

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Table 1

Relevant Safety Assessment Principles for Fault Studies - Containment and Severe Accidents - Considered During Step 4

SAP No.	SAP Title	Description
EKP.1	Engineering principles: key principles – Inherent safety	The underpinning safety aim for any nuclear facility should be an inherently safe design, consistent with the operational purposes of the facility.
EKP.2	Engineering principles: key principles – Fault tolerance	The sensitivity of the facility to potential faults should be minimised.
EKP.3	Engineering principles: key principles – Defence in depth	A nuclear facility should be so designed and operated that defence in depth against potentially significant faults or failures is achieved by the provision of several levels of protection.
ECS.4	ECS.4: Engineering principles: safety classification and standards – Codes and standards	For structures, systems and components that are important to safety, for which there are no appropriate established codes or standards, an approach derived from existing codes or standards for similar equipment, in applications with similar safety significance, may be applied.
ECS.5	Engineering principles: safety classification and standards – Use of experience, tests or analysis	In the absence of applicable or relevant codes and standards, the results of experience, tests, analysis, or a combination thereof, should be applied to demonstrate that the item will perform its safety function(s) to a level commensurate with its classification.
EDR.4	Engineering principles: design for reliability – Single failure criterion	During any normally permissible state of plant availability no single random failure, assumed to occur anywhere within the systems provided to secure a safety function, should prevent the performance of that safety function.
FA.1	Fault analysis: general – Design basis analysis, PSA and severe accident analysis	Fault analysis should be carried out comprising design basis analysis, suitable and sufficient PSA, and suitable and sufficient severe accident analysis.

Table 1

Relevant Safety Assessment Principles for Fault Studies - Containment and Severe Accidents - Considered During Step 4

SAP No.	SAP Title	Description
FA.2	Fault analysis: general – Identification of initiation faults	Fault analysis should identify all initiating faults having the potential to lead to any person receiving a significant dose of radiation, or to a significant quantity of radioactive material escaping from its designated place of residence or confinement.
FA.3	Fault analysis: general – Fault sequences	Fault sequences should be developed from the initiating faults and their potential consequences analysed.
FA.4	Fault analysis: general – Fault tolerance	DBA should be carried out to provide a robust demonstration of the fault tolerance of the engineering design and the effectiveness of the safety measures.
FA.9	Fault analysis: general – Further use of DBA	DBA should provide an input into the safety classification and the engineering requirements for systems, structures and components performing a safety function; the limits and conditions for safe operation; and the identification of requirements for operator actions.
FA.15	Fault analysis: severe accident analysis – Fault sequences	Fault sequences beyond the design basis that have the potential to lead to a severe accident should be analysed.
FA.16	Fault analysis: severe accident analysis – Uses of severe accident analysis	The severe accident analysis should be used in the consideration of further risk-reducing measures.
FA.17	Fault analysis: assurance of validity of data and models – Theoretical models	Theoretical models should adequately represent the facility and site.
FA.18	Fault analysis: assurance of validity of data and models – Calculation models	Calculational methods used for the analyses should adequately represent the physical and chemical processes taking place.
FA.19	Fault analysis: assurance of validity of data and models – Use of data	The data used in the analysis of safety-related aspects of plant performance should be shown to be valid for the circumstances by reference to established physical data, experiment or other appropriate means.

Table 1

Relevant Safety Assessment Principles for Fault Studies - Containment and Severe Accidents - Considered During Step 4

SAP No.	SAP Title	Description
FA.20	Fault analysis: assurance of validity of data and models – Computer models	Computer models and datasets used in support of the analysis should be developed, maintained and applied in accordance with appropriate quality assurance procedures.
SC.4	The regulatory assessment of safety cases – Safety case characteristics	In addition, Paragraph 93 of SC.4: requires demonstration that ALARP has been achieved for new facilities, modifications or periodic safety reviews, the safety case should: i) identify and document all the options considered, ii) provide evidence of the criteria used in decision making or option selection, and iii) support comparison of costs and benefits where quantified claims of gross disproportion have been made.

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Fault Studies Containment and Severe Accidents – UK EPR

Finding No.	Assessment Finding	Milestone (by which this item should be addressed)
AF-UKEPR-CSA-01	The licensee shall provide the ventilation strategy supporting the concept of inaccessible/accessible areas during normal operations and accident conditions for situations where one or more of the foils and dampers have failed.	Inactive commissioning – containment pressure test
AF-UKEPR-CSA-02	The licensee shall provide the test results to support the claims for the performance and the reliability of the foils and dampers used in the CONVECT system.	Inactive commissioning – containment pressure test
AF-UKEPR-CSA-03	The licensee shall provide clarification of the impact of the availability of the foils and dampers on plant operation and specifically, how this is controlled by technical specification.	Inactive commissioning – containment pressure test
AF-UKEPR-CSA-04	The licensee shall provide analysis to examine the impact of unintended and/or undetected opening of the foils and dampers on the pressure and temperature monitoring informing the accident management procedures.	Inactive commissioning – containment pressure test
AF-UKEPR-CSA-05	The licensee shall provide analysis to examine the impact of incomplete operation of the CONVECT system.	Inactive commissioning – containment pressure test
AF-UKEPR-CSA-06	The licensee shall justify that the isolation systems and containment penetrations meet the site specific loading requirements (pressure, temperature, moisture and leakage) in accident conditions.	Active commissioning – cold operations
AF-UKEPR-CSA-07	The licensee shall demonstrate that the design of insulation and the strainer structures associated with the safety injection system is such that the risk of sump blockage has been reduced to the lowest level reasonably practicable. In particular, the licensee should produce an analysis of the options and justify the choice of insulating technology.	Inactive commissioning – containment pressure test

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Fault Studies Containment and Severe Accidents – UK EPR

Finding No.	Assessment Finding	Milestone (by which this item should be addressed)
AF-UKEPR-CSA-08	The licensee shall justify the measurement systems indicating core conditions used to initiate the accident management procedures, such as, core outlet temperature measurements and the reliability of instrumentation routed via the RPV head; the justification should give consideration to common cause failure.	Active commissioning – cold operations
AF-UKEPR-CSA-09	The licensee shall provide an analysis of the impact on safety from degradation through ageing of the in-vessel thermocouples with a view of establishing maintenance plans assuring the integrity of this equipment over long operational periods and throughout the plant's lifetime.	Active commissioning – cold operations
AF-UKEPR-CSA-10	The licensee shall provide a robust justification of the operational requirements of the PDS during fault conditions. The justification is expected to fully consider the PDS implementation and Operating Strategies for Severe Accident (OSSA) for the UK EPR.	Active commissioning – cold operations
AF-UKEPR-CSA-11	The licensee shall provide validated evidence that either potential release of molten material from the reactor pit into the adjacent compartments is as low as reasonably practicable for the cases of central bottom RPV failure relating to late re-flood conditions, or that the melt release does not lead to the loss of containment integrity.	Construction – Nuclear island safety-related concrete
AF-UKEPR-CSA-12	The licensee shall provide an updated computational methodology to predict the MCCI progress within the reactor pit with a model of non isotropic ablation, supported by appropriate validation. This analysis could be performed by employing the existing COSACO model with different radial and axial heat flux efficiencies using values obtained from the 2D MCCI tests results.	Construction – Nuclear island safety-related concrete

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Fault Studies Containment and Severe Accidents – UK EPR

Finding No.	Assessment Finding	Milestone (by which this item should be addressed)
AF-UKEPR-CSA-13	The licensee shall demonstrate the presence of the layer inversion phenomenon for the bounding scenario of the minimum ablated concrete quantity. This justification is required to ensure that the risk associated with any significant interactions between water and the metallic layer is avoided. The response should also demonstrate that the resultant corium viscosity is appropriate for the bounding scenario of the maximum ablated concrete quantity.	Construction – Nuclear island safety-related concrete
AF-UKEPR-CSA-14	The licensee shall provide additional justification to: • demonstrate that the weld beads and outer frame meet the loading requirement, and • support a testing programme to capture unacceptable defects in the weld beads.	Construction – Nuclear island safety-related concrete
AF-UKEPR-CSA-15	The licensee shall justify that potential presence of chunks of concrete above the melt plug at the time of bottom head failure has no significant consequences on the melt plug opening.	Construction – Nuclear island safety-related concrete
AF-UKEPR-CSA-16	The licensee shall define the examination, maintenance, inspection and testing requirements necessary for the melt plug to fulfil its safety functions.	Inactive commissioning – containment pressure test
AF-UKEPR-CSA-17	The licensee shall provide the surveillance programme to monitor the zirconia stability for the plant lifetime.	Construction – Nuclear island safety-related concrete
AF-UKEPR-CSA-18	The licensee shall justify that suitable arrangements are in place to ensure that the IRWST water level is adequate for reasonably foreseeable faults.	Construction – Nuclear island safety-related concrete
AF-UKEPR-CSA-19	The licensee shall demonstrate the level of blockage in the channels, under the cooling plates, that can be tolerated before their safety function is impaired.	Construction – Nuclear island safety-related concrete
AF-UKEPR-CSA-20	The licensee shall provide updated spreading calculations for bounding scenarios employing appropriate viscosities and melt plug opening cross sectional areas.	Construction – Nuclear island safety-related concrete

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Fault Studies Containment and Severe Accidents – UK EPR

Finding No.	Assessment Finding	Milestone (by which this item should be addressed)
AF-UKEPR-CSA-21	The licensee shall provide the measure(s) and arrangement(s) for inspection in order to ensure that the reactor pit is kept sufficiently dry.	Inactive commissioning – containment pressure test
AF-UKEPR-CSA-22	The licensee shall provide a comprehensive examination of re-criticality for all reasonably foreseeable conditions during the transient progression and within the CMSS.	Construction – Nuclear island safety-related concrete
AF-UKEPR-CSA-23	The licensee shall justify that the measures taken to mitigate hydrogen-related risk set out in response to RO-UKEPR-78 are ALARP and, in particular, that there are no reasonably practical measures that would increase mixing of hydrogen plumes during a delayed depressurisation of the primary circuit in a severe accident.	Construction – Nuclear island safety-related concrete
AF-UKEPR-CSA-24	There are a number of observations made with regards to the operational requirements for PARs during accident scenarios. Given the significance of the equipment to hydrogen concentration management during accident progression, the licensee shall provide additional justification that; • considers the poisoning of the PARs by the released fission products informed by the outcomes of the planned experimental programme, and • demonstrates the continued operability of the PARs in prolonged accident scenarios.	Construction – Nuclear island safety-related concrete
AF-UKEPR-CSA-25	The licensee shall provide the available measures to limit the containment pressure, in the event of a severe accident leading to the failure of the CHRS, to prevent uncontrolled radiological releases from the primary containment.	Construction – Nuclear island safety-related concrete

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Fault Studies Containment and Severe Accidents – UK EPR

Finding No.	Assessment Finding	Milestone (by which this item should be addressed)
AF-UKEPR-CSA-26	The licensee shall provide a comprehensive set of documentation for the GASFLOW and the COM3D codes used in support of the PCSR. This should include, but not be restricted to: • Detailing the modelling used, • Guidance on the code limits of applicability, its use and qualified uncertainty allowances, and • Substantiation of the codes' validity by comparison against measurements and independent analysis.	Construction – Nuclear island safety-related concrete

Note: It is the responsibility of the Licensees / Operators to have adequate arrangements to address the Assessment Findings. Future Licensees / Operators can adopt alternative means to those indicated in the findings which give an equivalent level of safety.

For Assessment Findings relevant to the operational phase of the reactor, the Licensees / Operators must adequately address the findings <u>during</u> the operational phase. For other Assessment Findings, it is the regulators' expectation that the findings are adequately addressed no later than the milestones indicated above.

Annex 2

GDA Issues – Fault Studies – Containment and Severe Accidents – UK EPR

There are no GDA Issues for this topic area.