Office for Nuclear Regulation
An agency of HSE

Generic Design Assessment – New Civil Reactor Build

GDA Close-out for the EDF and AREVA UK EPR™ Reactor
GDA Issue GI-UKEPR-IH-02 Revision 2 – Verification and Validation

Assessment Report: ONR-GDA-AR-12-017
Revision 0
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EXECUTIVE SUMMARY

This report presents the close-out of the Office for Nuclear Regulation's (an agency of HSE) Generic Design Assessment (GDA) for the GDA Issue **GI-UKEPR-IH-02 Revision 2** and the associated GDA Issue Actions generated as a result of the GDA Step 4 Internal Hazards Assessment of the UK EPR™. The assessment has focussed on the deliverables identified within the EDF and AREVA Resolution Plan published in response to the GDA Issue and on further assessment undertaken of those deliverables.

During Step 4 evidence was sought relating to the verification and validation for a number of internal hazards. In a number of cases, this information was not available as the work had yet to be undertaken for Flamanville 3 (FA3). Given the sampling approach adopted by ONR in undertaking assessment, the requisite verification and validation was requested through this GDA Issue.

The approach taken by EDF and AREVA was to provide the verification and validation studies that have been undertaken for Flamanville 3 in the areas of internal flooding, protected cable routes, high energy line break, and internal missile. These studies were included within the Resolution Plan provided for this GDA Issue by EDF and AREVA.

Further to the receipt of the deliverables detailed within the Resolution Plan together with the responses to the Technical Queries (TQ) raised, I am satisfied that the approach taken to the verification and validation for the internal hazards in the areas of internal flooding, cable routing, high energy line break, and internal missile has been shown to be to a good standard.

My judgement of the adequacy of the response to the GDA issue is based upon the following factors:

- The approach to the analyses provided has been comprehensive and robust for each of the areas in which verification and validation was sought.
- The analyses themselves have demonstrated that the potential for loss of more than one redundancy is low, and where the potential exists for more than one redundancy to be threatened there are robust operational arguments presented to demonstrate that loss of more than one redundancy would be tolerable, and nuclear safety would not be compromised. In some cases, diverse means by which nuclear safety can be assured have been presented as part of the analysis.
- The submissions provided in response to the GDA issue together with the PCSR have shown to align with my expectations in relation to standards, guidance, and relevant good practice.
- There have been 4 assessment findings raised as a result of my assessment, which are largely due to the need for the future licensee to capture the findings and approach taken for the verification and validation that has been undertaken for FA3 and address aspects of the findings presented.

I am, therefore, satisfied that GDA Issue, **GI-UKEPR-IH-02**, can now be closed.

LIST OF ABBREVIATIONS

ALARP	As low as is reasonably practicable
AREVA	AREVA NP SAS
C&I	Control and Instrumentation
CCWS	Component Cooling Water System
CHRS	Containment Heat Removal System
CMF	Change Modification Form
CSBVS	Controlled Safeguard Building Ventilation System
CVCS	Chemical and Volume Control System
DAC	Design Acceptance Confirmation
DB	Diesel Building
DCL	Control and Electrical Room Atmospheric Conditioning
DEL	Safety Chilled Water System
DER	Reactor Building Chilled Water System
DFL	Smoke Control System
DWL	Safeguard Building Ventilation System
EBS	Extra Borating System
EDF	Electricité de France SA
EFWS	Emergency Feedwater System
EVF	Reactor Building Internal Filtration System
EVR	Containment Cooling Ventilation System
FA3	Flamanville 3 Nuclear Power Station
FB	Fuel Building
FPCS	Fuel Pool Cooling System
GDA	Generic Design Assessment
HELB	High Energy Line Break
HIC	High Integrity Components
HP	High Pressure
HRA	Reactor Building Containment
HRB	Reactor Building Annulus
HSE	Health and Safety Executive
IAEA	International Atomic Energy Agency
IRWST	In-containment Refuelling Water Storage Tank
JPI	Nuclear Island Fire Protection System

LIST OF ABBREVIATIONS

JPV	Diesel Building Fire Protection and Fire-fighting Distribution System
LHSI	Low Head Safety Injection
LOCA	Loss of Coolant Accident
LUHS	Loss of Ultimate Heat Sink
MCR	Main Control Room
MFWS	Main Feedwater System
MHSI	Medium Head Safety Injection
MSSS	Main Steam System
NAB	Nuclear Auxiliary Building
NI	Nuclear Island
NSS	Nuclear Sampling System
NVDS	Nuclear Vent and Drain System
ONR	Office for Nuclear Regulation (an agency of HSE)
PCC	Plant Condition Category
PCSR	Pre-construction Safety Report
PDMS	Plant Design and Modelling System
RB	Reactor Building
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RES	Steam Generator Secondary Side Sampling System
RHRS	Residual Heat Removal System
RPE	Nuclear Island Vent and Drainage System
RSS	Remote Shutdown Station
SAB	Safeguard Auxiliary Building
SAP	HSE Safety Assessment Principles
SAT	Service Compressed Air Distribution System
SBO	Station Black-Out
SED	Demineralised Water Distribution System
SEP	Drinking Water System
SFA	Access Fire Compartment
SGBS	Steam Generator Blowdown System
SIS	Safety Injection System
SSCs	Systems, Structures and Components

LIST OF ABBREVIATIONS

SSSS	Standstill Seal System
TAG	Technical Assessment Guide(s) (ONR)
TLOCC	Total Loss of Cooling Chain
TQ	Technical Query
UK EPR™	EDF and AREVA UK specific pressurised water reactor design
WENRA	Western European Nuclear Regulators' Association

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1 INTRODUCTION

1.1 Background

1 This report presents the close-out of the Office for Nuclear Regulation's (an agency of HSE) Generic Design Assessment (GDA) for the GDA Issue **GI-UKEPR-IH-02 Revision 2** and the associated GDA Issue Actions (Ref. 6) generated as a result of the GDA Step 4 Internal Hazards Assessment of the UK EPR™ (Ref. 7). The assessment has focussed on the deliverables identified within the EDF and AREVA Resolution Plans (Ref. 8) published in response to the GDA Issue and on further assessment undertaken of those deliverables.

2 GDA followed a step-wise-approach in a claims-argument-evidence hierarchy. In Step 2 the claims made by EDF and AREVA were examined, and in Step 3 the arguments that underpin those claims were examined. The Step 4 assessment reviewed the safety aspects of the UK EPR™ reactor in greater detail by examining the evidence supporting the claims and arguments made in the safety documentation.

3 The Step 4 Internal Hazards Assessment identified a number of GDA Issues and assessment findings as part of the assessment of the evidence associated with the UK EPR™ reactor design. A GDA Issue is an observation of particular significance that requires resolution before the Office for Nuclear Regulation (ONR), an agency of HSE, would agree to the commencement of nuclear safety related construction of the UK EPR™ within the UK. An assessment finding results from a lack of detailed information which has limited the extent of assessment and as a result the information is required to underpin the assessment. However, these assessment findings are to be carried forward as part of normal regulatory business.

4 The Step 4 Assessment concluded that the UK EPR™ reactor was suitable for construction in the UK subject to resolution of 31 GDA Issues. The purpose of this report is to provide the assessment which underpins the judgement made in closing GDA Issue **GI-UKEPR-IH-02**.

1.2 Scope

5 This report presents only the assessment undertaken as part of the resolution of this GDA Issue and it is recommended that this report be read in conjunction with the Step 4 Internal Hazards Assessment of the EDF and AREVA UK EPR™ in order to appreciate the totality of the assessment of the evidence undertaken as part of the GDA process.

6 This assessment report is not intended to revisit aspects of assessment already undertaken and confirmed as being adequate during previous stages of the GDA. However, should evidence from the assessment of EDF and AREVA's responses to GDA Issue highlight shortfalls not previously identified during Step 4, there will be a need for these aspects of the assessment to be highlighted and addressed as part of the close-out phase, or be identified as assessment findings to be taken forward to the site specific phase.

7 The possibility of further assessment findings being generated as a result of this assessment is not precluded given that resolution of the GDA Issue may leave aspects of the assessment requiring further detailed evidence when the information becomes available at a later stage.

8 During Step 4 (Ref. 7), evidence was sought relating to the verification and validation for a number of internal hazards. In a number of cases, this information was not available as the work had yet to be undertaken for Flamanville 3 (FA3). Given the sampling approach

adopted by ONR in undertaking assessment, the requisite verification and validation was requested through a GDA Issue which stated:

"Outstanding Verification and Validation for internal flooding, cable routing, high energy line break and missiles forms part of the requisite evidence and will be required in order to demonstrate an adequate internal hazards safety case."

9 As part of this GDA Issue, four GDA Issue Actions were raised associated with each of the internal hazards areas stated within the GDA Issue. The actions sought further arguments and evidence to support the claims made within the case through requesting detailed analysis of the potential hazards, including detailed claims on SSCs that had not previously been specifically captured within the March 2011 Consolidated PCSR (Ref. 11) e.g. claims on barriers and doors against the effects of internal flooding, the routing and identification of fire protected cable trays, claims made on barriers and pressure relief systems associated with High Energy Line Break (HELB), etc.

10 ONR suggested that the requisite evidence could be undertaken through the provision of verification and validation reports for Flamanville 3 (FA3) and should include consideration of the following:

- Internal Flooding
  - Civil structures (including surface coatings) claimed as flood barriers.
  - Watertight doors and penetrations including qualification data.
  - Drains and sumps claimed to prevent damage to nuclear significant SSCs.
  - Calculations in place to support any claims made on potential water volumes.
  - Any further defence in depth and ALARP measures that could be implemented into the design.
  - Any identified design changes and their implementation within the PCSR.
- Routing and Fire Protection of Electrical Cable Trays.
  - The routing and identification of protected cable trays.
  - Justification of claims and arguments made relating to geographical separation.
  - The provision of passive protection applied to cables and cable trays specifically.
  - Any further defence in depth and ALARP measures that could be implemented into the design.
  - Any identified design changes and their implementation within the PCSR.
- High Energy Line Break
  - Consequence analysis, where applicable.
  - Break preclusion.
  - Identification and qualification of physical restraints, barriers and doors.
  - Identification and qualification of pressure relief panels/routes.
  - Any further defence in depth and ALARP measures that could be implemented into the design.
  - Any identified design changes and their implementation within the PCSR.
- Internal Missile

- Identification of potential sources of internal missile which could result in a threat to nuclear safety significant SSCs.
- Consequence analysis, where applicable.
- Break preclusion.
- Identification and qualification of physical restraints, barriers and doors.
- Any further defence in depth and ALARP measures that could be implemented into the design.
- Any identified design changes and their implementation within the PCSR.

1.3 Methodology

11 The methodology applied to this assessment is identical to the approach taken during Step 4 which followed the ONR HOW2 document PI/FWD, *"Permissioning – Purpose and Scope of Permissioning"* (Ref. 1), in relation to mechanics of assessment within ONR.

12 This assessment has been focussed primarily on the submissions relating to resolution of the GDA Issue as well as any further requests for information or justification derived from assessment of those specific deliverables.

13 The aim of this assessment is to provide a comprehensive assessment of the submissions provided in response to the GDA Issue to enable ONR to gain confidence that the concerns raised have been resolved sufficient that they can either be closed or lesser safety significant aspects be carried forward as assessment findings.

1.4 Structure

14 This Assessment Report structure differs slightly from the structure adopted for the previous reports produced within GDA, most notably the Step 4 Internal Hazards Assessment. The report has been structured to reflect the assessment of the individual GDA Issue rather than a report detailing close-out of all GDA Issues associated with this technical area.

15 The reasoning behind adopting this report structure is to allow closure of GDA Issues as the work is completed rather than having to wait for the completion of all the GDA work in this technical area.

2 ONR'S ASSESSMENT STRATEGY FOR INTERNAL HAZARDS

16 The intended assessment strategy for GDA Close-out for the internal hazards topic area was set out in an Assessment Plan (Ref. 12) that identified the intended scope of the assessment and the standards and criteria that would be applied.

17 The overall bases for the assessment of the GDA Issue are the internal hazards elements of:

- Submissions made to ONR in accordance with the resolution plans.
- Update to the Submission / Pre-construction Safety Report (PCSR) / Supporting Documentation.
- The Design Reference that relates to the Submission / PCSR as set out in UK EPR™ GDA Project Instruction UKEPR-I-002 (Ref. 9) which will be updated throughout GDA Issue resolution. This includes Change Management Forms (CMF) agreed for inclusion within GDA.

2.1 The Approach to Assessment for GDA Close-out

18 The approach to the closure of GDA Issues for the UK EPR™ Project involves:

- Assessment of submissions made by EDF and AREVA in response to GDA Issues identified through the GDA process. These submissions are detailed within the EDF and AREVA Resolution Plan for the GDA Issue. In the event of requiring further supporting evidence for the assessment, Technical Queries (TQ) (Ref. 14) have been generated.
- The objective of the Internal Hazards Assessment has been to assess submissions made by EDF and AREVA in response to the GDA Issues identified through the GDA process and the design changes requested by EDF and AREVA and, if judged acceptable, clear the GDA Issues.

2.2 Standards and Criteria

19 The relevant standards and criteria adopted within this assessment are principally the Safety Assessment Principles (SAPs) (Ref. 2), internal ONR Technical Assessment Guides (TAG) (Ref. 3), relevant national and international standards and relevant good practice informed from existing practices adopted on UK nuclear licensed sites. The key SAPs and relevant TAGs have been detailed within this section. National and international standards and guidance have been referenced where appropriate within the assessment report. Relevant good practice, where applicable, has also been cited within the body of the assessment.

2.2.1 Safety Assessment Principles

20 The key SAPs (Ref. 2) applied within the Internal Hazards Assessment of the EDF and AREVA UK EPR™ are included within Table 1 of this report.

2.2.2 Technical Assessment Guides

21 The following Technical Assessment Guides have been used as part of this assessment (Ref. 3):

- T/AST/006 Issue 03 - Deterministic Safety Analysis and the Use of Engineering Principles in Safety Assessment.

- T/AST/014 Issue 02 - Internal Hazards.
- T/AST/017 Issue 02 - Structural Integrity Civil Engineering Aspects.
- T/AST/036 Issue 02 – Diversity, Redundancy, Segregation and Layout of Mechanical Plant.
- T/AST/051 Issue 01 – Guidance on the Purpose, Scope and Content of Nuclear Safety Cases.

2.2.3 International Standards and Guidance

22 The following international standards and guidance have been used as part of this assessment:

- Western European Nuclear Regulators' Association. Reactor Harmonization Group. WENRA Reactor Reference Safety Levels (Ref. 4).
- Safety of Nuclear Power Plants: Design. Safety Requirements, NS-R-1(Ref. 5)
- Protection against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants. Safety Guide, NS-G11.11 (Ref. 5)
- Protection against Fires and Explosions in the Design of Nuclear Power Plants. Safety Guide, NS-G-1.7 (Ref. 5)

2.3 Use of Technical Support Contractors

23 No Technical Support Contractors were utilised in the assessment of this GDA Issue.

2.4 Out-of-scope Items

24 As part of the GDA Closeout, no items have been identified as being out of scope by EDF and AREVA as a result of this assessment.

3 EDF AND AREVA DELIVERABLES IN RESPONSE TO THE GDA ISSUE

25 In response to the GDA Issue, EDF and AREVA provided a Resolution Plan (Ref. 8) detailing how they intended to address the above points. The Resolution Plan stated that:

- **GI-UKEPR-IH-02.A1** - Internal Flooding – *"EDF/ AREVA will transmit documents from the Flamanville 3 EPR studies to support the internal flooding safety case. As they form the basis of the UK EPR™ design they are applicable to the UK EPR™. The dedicated internal flooding safety case will be provided during the Nuclear Site Licensing (NSL) Phase."*
- **GI-UKEPR-IH-02.A2** - Cable Routing – *"To support the fire hazard case, EDF/AREVA will transmit a document from the Flamanville 3 EPR studies that will detail the routing of cable trays, identify foreign division cable trays and give details on cable trays protected against fire. As they form the basis of the UK EPR™ design they are applicable to the UK EPR™. The dedicated internal fire hazard case will be provided during the NSL Phase."*
- **GI-UKEPR-IH-02.A3** – High Energy Line Break – *"EDF/AREVA will transmit additional documents from the Flamanville 3 EPR project to support the High Energy Line Break case. As they form the basis of the UK EPR™ design they are applicable to the UK EPR™. The dedicated high energy line break case will be provided during the NSL Phase."*
- **GI-UKEPR-IH-02.A4** – Internal Missile – *"EDF/AREVA will transmit a document from the Flamanville 3 EPR project to support the Internal Missile case. This case is based on the claim that RCC-M components are not potential sources of missiles. The content of the transmitted case is based on the detailed analysis of the non-RCCM components. As they form the basis of the UK EPR design, the Flamanville 3 EPR studies on non- RCC-M components are applicable to the UK EPR. The dedicated UK EPR internal missile case on these components will be provided during the NSL phase.*

  A dedicated internal missile case on some RCC-M components which are not identified as High Integrity Components (HIC) will be developed in the frame of the resolution of the GDA issue GIUKEPR- IH-04."

26 The information provided by EDF and AREVA in response to this GDA Issue was broken down into the following specific deliverables for detailed assessment:

GDA Issue Action	Internal Hazards Area	Deliverable	Overview within Section	Assessment within Section
GI-UKEPR-IH-02.A1	Internal Flooding	HN flooding analysis: "Flooding Safety Analysis for Nuclear Auxiliary Building HNX", PF2009EN0001 Revision D (Ref. 15)	3.1.1	4.2.1.1
GI-UKEPR-IH-02.A1	Internal Flooding	SAB flooding analysis: "Flooding event – Analysis of the Safeguard Building", EZT2009EN0005 Revision E (Ref. 16).	3.1.2	4.2.1.2
GI-UKEPR-IH-02.A1	Internal Flooding	EPR FA3 Fuel Building Internal Flooding Assessment, EYRT2009FR0008 Revision C (Ref. 17).	3.1.3	4.2.1.3
GI-UKEPR-	Internal Flooding	Study of Internal Flooding Events in the Diesel Generator Building of the Flamanville 3 EPR.	3.1.4	4.2.1.4

GDA Issue Action	Internal Hazards Area	Deliverable	Overview within Section	Assessment within Section
IH-02.A1		EYRT2009FR0032 Revision E (Ref. 18).		
GI-UKEPR-IH-02.A1	Internal Flooding	Analysis of internal flooding in the FA3 EPR reactor building, EYRT2009FR0076 Revision D (Ref. 19).	3.1.5	4.2.1.5
GI-UKEPR-IH-02.A2	Identification and Location of Protected Cable Trays.	Location of cable trays in the FA3 EPR Nuclear Island, EYRT2010FR0042 Revision B (Ref. 20).	3.2.1	4.2.2.1
GI-UKEPR-IH-02.A3	High Energy Line Break (HELB) Studies	Application of the MTE175 "Breaches of High Energy Pipes, load caused on the structural steelworks" ECEIG091393 Revision A1 (Ref. 21).	3.3.1	4.2.3.1
GI-UKEPR-IH-02.A3	High Energy Line Break (HELB) Studies	Additional information for the stage 1 analysis report on the consequences of high energy line breaks – fuel building, ECEF101595 Revision A1 (Ref. 22).	3.3.2	4.2.3.2
GI-UKEPR-IH-02.A3	High Energy Line Break (HELB) Studies	Additional information for the stage 1 analysis report on the consequences of high energy line breaks – safeguard auxiliary buildings, ECEF101596 Revision A1 (Ref. 23).	3.3.3	4.2.3.3
GI-UKEPR-IH-02.A4	Internal Missile	EPR Internal Missiles – Risk assessment report on building structure and layout. ECEIG091634, Revision B1 (Ref. 24).	3.4.1	4.2.4.1

27 An overview of each of the deliverables is provided within this section. It is important to note that this information is supplementary to the information provided within the March 2011 Consolidated PCSR (Ref. 11) which has already been subject to assessment during earlier stages of GDA. In addition, it is important to note that the deliverables are not intended to provide the complete safety case for the verification and validation for each of the hazards detailed within this assessment. Rather they form further detailed arguments and evidence to supplement those already provided during earlier Steps within the GDA Process.

28 The deliverables associated with this GDA Issue use the existing French approach to classification and categorisation of Structures, Systems, and Components (SSCs). This has been identified as requiring resolution through the issue of an assessment finding as part of the work undertaken in response to the cross cutting GDA Issue, **GI-UKEPR-CC-01**.

3.1 EDF and AREVA Submissions Relating to Internal Flooding Verification and Validation, GI-UKEPR-IH-02.A1

29 For each of the analyses undertaken, the safety objectives are identical in that an internal flooding event should not result in loss of more than one F1 redundancy nor should there be any propagation of flooding between safety classified buildings. An F1 function is that which is required to either attain a controlled shutdown state (F1A) or to secure safe shutdown after the controlled state has been reached (F1B). In order to ensure that such claims can be substantiated, there are claims made on civil structures to be watertight as

well as the identification and location of main internal flooding initiators. The analyses also consider the flow of any potential internal flooding event and consider drainage routes including engineered drains, doors, and stairways.

30 In addition, there are a number of assumptions derived from the March 2011 Consolidated PCSR (Ref. 11), namely:

- If the leak/break can be detected by the Control and Instrumentation (C&I) systems and if automatic isolation is implemented, the duration of the release is the time taken to detect it plus the time taken to automatically isolate it.
- If the leak/break can be detected by signals in the Main Control Room (MCR) and if it is designed to be isolated by manual action in the MCR, the release duration is the time for the first alarm to appear in the MCR plus a 30 minute time allowance for manual action to be performed in the MCR.
- If the leak/break can be detected by signals in the MCR and it is designed to be isolated by on-site field action, the release duration is the time for the first alarm to appear in the MCR plus the time for personnel to perform field action such as manually isolating a valve. The time required for on-site field action is assumed to be 1 hour.
- If the leak/break cannot be detected or if isolation is not possible, release of the entire inventory is assumed if the flow is not limited in some other way.

31 The following sections provide an overview of the EDF and AREVA submissions relating to the specific analyses that have been undertaken for each of the Nuclear Island (NI) buildings.

3.1.1 Flooding Safety Analysis for Nuclear Auxiliary Building, PF2009EN0001. Revision D

32 The above submission (Ref. 15) provides the further analysis associated with internal flooding of the Nuclear Auxiliary Building (NAB) for the reference Flamanville (FA3) design. The analysis serves to verify that the an internal flooding event in a safety related area does not include a risk of common cause failure of redundant safety classified SSCs with respect to achieving the safety objectives of safe shutdown, residual heat removal, limitation of radiological consequences and assuring the functions of supporting safety classified SSCs.

33 The report identifies that there are no systems contained within the NAB that are required to achieve a safe shutdown state. The report focuses on the potential for pipe failure within the NAB and the need to contain the water within the bounds of the building in order to ensure containment of any potential radioactivity. As part of this, consideration is given to flooding of other buildings including the technical galleries as well as to the environment arising from earthquakes.

34 The report concludes that all potential flooding scenarios arising from both single pipe failure as well as a result of multiple pipework failures can be adequately contained within the NAB beneath 0.00m level such that there is no effect on achieving a safe shutdown state and no release of activity to the environment.

3.1.2 Flooding event – Analysis of the Safeguard Building, EZT2009EN0005, Revision E

35 There is detailed analysis included within the Safeguard Auxiliary Building (SAB) flooding analysis (Ref. 16) which considers internal flooding events arising from pipework failures which could result in loss of more than one redundancy. The analysis considers that loss of divisional separation below 0.00m level is not possible as one redundancy is contained

within each of the buildings and separated utilising the claimed [redacted] metre barriers. The analysis does identify a number of areas, principally above the 0.00m level to which the internal flood barriers do not extend above where there are challenges to the principle of divisional separation, namely:

- The valve compartments in Divisions 1 and 4.
- The Main Control Room (MCR).
- The Remote Shutdown Station (RSS).
- The control room and electrical equipment rooms atmospheric conditioning (DCL) trains in Divisions 2 and 3.
- F1 redundancies located in the same compartment.

36 As part of the analysis, consideration has been given to water drainage and interfaces within the buildings at levels above 0.00m, including doors, ductwork and piping penetrations.

37 The analysis does identify a number of modifications to plant and layout to ensure that flooding induced by pipework failure does not threaten the adequacy of the design to withstand internal flooding events. The following modifications have been identified that are necessary to ensure that an internal flooding event does not affect more than one division.

- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted]

38 The analysis concludes:

- The UK EPR™ design has sufficient provision in place to ensure that water drains to the lowest parts of the building and that there is sufficient capacity in which to store any postulated internal flooding event.
- Redundant F1 components are sufficiently protected and in exceptional cases, where there could be loss of redundant F1 components, the analysis demonstrates that such loss arising from internal flooding is acceptable.
- The ability to manage PCC transients is unaffected by internal flooding events. Initiations of PCC3/4 results only from the break assumptions in the flooding analysis, which postulate ruptures that are themselves PCC3/4 events.

3.1.3 EPR FA3 Fuel Building Internal Flooding Assessment, EYRT2009FR008, Revision C

39 The analysis detailed within the above submission (Ref. 17) considers internal flooding resulting in loss of more than F1 redundancy and claims the barriers between the Fuel Building and other safety classified buildings to a water column height of [redacted]. In

addition, the segregation barrier between each half of the Fuel Building (FB) is claimed to provide segregation to the same 10 metre water column height in order to ensure that loss of more than one F1 redundancy is prevented.

40 For flooding above 0.00m level there are claims made on the doors across the divisional barriers to ensure that more than one F1 redundancy is not lost. There are also specific requirements for two removable concrete slabs located within Room [redacted] associated with the Chemical and Volume Control System (CVCS) system that allows removal of the pumps. Removal of these slabs links both Divisions 1 and 4 thus the requirements for the slabs and the area in general are:

- The slabs are to be provided with watertight seals as are any penetrations provided within the floor.
- The walls are to be watertight to a height of [redacted] as are the doors to the room.
- Both slabs are not to be removed at the same time.
- An isolation valve is to be installed to isolate the floor drain in [redacted], which is to be closed when the slab to the Division 4 half of the FB is removed. During all other times it is left open to ensure that any flood water passes into the Division 1 half of the FB.

41 The analysis also provides details of all the doors, walls, and floor slabs required to be claimed against the effects of flooding and specifies the maximum head of water it must contain. This is done for divisional segregation between Divisions 1 and 4 of the FB as well as between it and other safety classified buildings. Included within the analysis is the need to ensure that penetrations provided in either the horizontal or vertical barriers are adequately sealed against the potential water column height also.

42 There is consideration of the impact of internal flooding on F1 redundancies which focuses on SSCs in locations within the same division. The analysis undertakes an F1 redundancy impact assessment of the areas and identifies any further measures required to ensure that loss of more than one redundant F1 system is prevented, including:

- [redacted]
- [redacted]
- [redacted]

43 The analysis includes figures illustrating the location of claimed barriers and doors within FB as well as the supporting analysis for flow rates including room by room flood depths and the claim detection and isolation systems including their classification.

44 The analysis within Reference 17 states that the following measures should be implemented to ensure that the safety objectives associated with preventing loss of more than one F1 redundancy and flooding spread to adjacent safety classified buildings can be met:

- *"Take benefit of design quality of Q3 pipes (see Appendix F)*
- *Use the detection and isolation devices mentioned in Appendix D*

- *Reclassify as "F2 Operable" all currently non-classified detection and isolation devices mentioned in Appendix D*
- *Implement changes listed in §8.1 intended to prevent any common mode risk involving the opening of the removable floor slabs in room* [redacted]
- *Comply with the door leak tightness requirements specified in Table 20*
- *Comply with the wall and floor slab leak tightness requirements specified in Tables 21, 22 and 23*
- *Qualify as spray-resistant the remote electronics units for measurements* [redacted] *in room* [redacted] *(see §9.3.2), or install them in sealed boxes*
- *Implement the changes to the nuclear island demineralised water distribution system (SED) and the nuclear island fire protection system (JPI) explained in engineering change notice CSFL0668*
- *Install a 10cm high concrete wall around common shaft* [redacted] *(see §9.3.8)*
- *Fit seals underneath fire doors such that their orientation allows water to flow most freely in the direction shown on the drawings in Appendix A*
- *Install a shield to protect pump* [redacted] *against water jets (see §9.6)*
- *Conduct a functional analysis demonstrating the acceptability of the loss of the CVCS redundancies on level* [redacted]*m"*

45 The analysis concludes that the safety objectives are met providing the above measures are implemented within the design.

3.1.4 Study of Internal Flooding Events in the Diesel Generator Building of the Flamanville 3 EPR, EYRT2009FR0032 , Revision E

46 The analysis detailed within the above submission (Ref. 18) considers internal flooding resulting in loss of more than F1 redundancy and principally claims the barriers between the divisions of the Diesel Buildings (DB) throughout the full height of the building from the [redacted]m level upwards. The analysis of internal flooding is limited to one of the two diesel buildings given their identical design. The analysis focuses on the divisional segregation between the two divisions of backup diesel supply together with the Station Blackout Diesels (SBO) as well as the need to prevent water ingress into adjacent areas, namely, the technical galleries.

47 There are five fluid containing systems which have been subject to analysis within the submission, namely:

- The Diesel Building fire fighting system (JPV).
- The Nuclear Island fire fighting hydrant system (JPI) which supplies the JPV system.
- The drinking water system (SEP)
- The demineralised water system (SED).
- Fuel tanks for the main diesel generators and station black-out (SBO) diesels.

48 Each of the systems has been considered within the submission and the potential flood heights, flow rates, and retention volumes analysed. As part of the analysis, a room by

room study has been completed to confirm that, for the postulated flooding events, it is not possible for flooding to spread to another division or building.

49 As with the other internal flooding analyses undertaken, the outcome of the room by room analysis involved the consideration of watertight/fluid tight barriers, drainage routes, and detection and isolation measures. Detailed consideration has also been given to the risk of flood propagation through openings and sleeves with the identification of those which could be submerged in the event of a flooding event. In the event where an opening of a sleeve could be submerged the amount by which it is submerged is then claimed as being the required leak tightness for that particular opening or sleeve.

50 In addition to the analysis undertaken within the DBs, there is consideration of both flooding arising to and from the adjoining buildings. The only buildings that adjoin the DBs are the Technical Galleries and the analysis shows that the maximum amount of water that could be released into the galleries is [redacted]. However, the potential flow from the Technical Galleries could be significantly greater due to failures in the systems within those areas. It is noted that water flow into and from adjoining buildings is not acceptable from a safety case perspective as it does not meet the safety requirements set out within the PCSR (Ref. 11) and as a result the need to ensure that the doors, HDA0603DO and HDB0603DO, are designed to be watertight has been identified. The report identifies that the particular doors are “anti-panic” doors and that the designer cannot guarantee the feasibility of both leak-tightness and “anti-panic” requirements.

51 The report concludes that the three distinct divisions (two emergency diesels and one SBO diesel) of the Diesel Buildings demonstrate that one division has no impact on the other, providing the penetrations through the barrier wall are suitably protected for the head of water claimed. As a result there are no initiators within the Diesel Buildings that could result in a common mode failure of F1 systems.

52 It recognises that there are preventative isolations undertaken for the SED and SEP systems to limit the volume of water released on detection within the sumps contained within the DBs.

53 The submission concludes:

“The design of the three distinct divisions of the diesel generator buildings allow to demonstrate that one division has no impact on another division insofar as plugging material of the openings support the heads of water calculated in Section 7 [of the submission]. *Therefore, provided this condition is fulfilled, there is no common mode risk.”*

54 In the case of flooding from the Diesel Buildings into the adjoining Technical Galleries, the principle associated with the prevention of flooding to a classified building cannot be met, however, the submission states, *“the flooding scenario is covered by the break scenarios for these same JPI and SED pipes within the galleries”.*

3.1.5 Analysis of Internal Flooding within the EPR FA3 Reactor Building, EYRT2009FR0076, Revision D

55 The analysis detailed within the above submission (Ref. 19) determines the potential water volume arising from an internal flooding event and then identifies the F1 safety classified equipment that could be impacted. The flooding events are assumed to occur during normal operation, however, Plant Condition Categories (PCC), PCC-3 and PCC-4 events resulting in flooding are considered as part of the accident analysis work undertaken within Chapter 14 of the PCSR (Ref. 11). PCC-3 and PCC-4 events are defined within the PCSR as:

- PCC-3 events include all design basis incidents, characterised by initiating events with an estimated frequency of occurrence within the range of 10^{-2} to 10^{-4} per year.
- PCC-4 events include all design basis incidents, characterised by initiating events with an estimated frequency of occurrence within the range of 10^{-4} to 10^{-6} per year.

56 Flooding associated with spurious or inadvertent operation of fire protection systems is also addressed within the analysis undertaken.

3.1.5.1 Flooding of the Reactor Building Containment (HRA)

57 One of the key methods by which to detect flooding within the Containment are the redundant level sensors, [redacted] located within the In-Containment Refuelling Water Storage Tank (IRWST). This ensures that any flooding event will be detected no later than the high level of the IRWST. Each of the level monitors are safety class F1B and are installed at an elevation of [redacted] which corresponds to the maximum level of the IRWST in normal operation. On detection, alarm [redacted] is triggered within the Main Control Room (MCR).

58 The analysis identifies two potential scenarios for flooding within the Containment either through engineered routes to the IRWST or contained within rooms at which point equipment contained within the room is assumed to be submerged, named “dead volumes” within the submission.

59 In the case of overfilling of the IRWST, the scenario considers that the initial level in the tank is at its highest level when the flooding event occurs. At that point the flood water overflows and floods the floors above it.

60 For the scenario involving “dead volumes”, each room is identified and the submission states that there is no F1 safety class equipment located within them.

61 In addition, there are rooms that have been specifically designed to retain the water volumes and are included within the analysis undertaken. Ultimately, these volumes eventually flow into the IRWST.

62 Detailed analysis of the water retaining systems within Containment are included within the submission:

- Steam Generator Blowdown System (SGBS)
- Main Feedwater System (MFWS)
- Emergency Feedwater System (EFWS)
- Reactor Building Chilled Water System (DER)
- Reactor Building Internal Filtration System (EVF)
- Containment Cooling Ventilation System (EVR)
- Containment Heat Removal System (CHRS)
- Nuclear Island Fire Protection System (JPI)
- Fuel Pool Cooling (and Purification) System (FPCS)
- Extra Borating System (EBS)
- Reactor Coolant System (RCS)
- Chemical and Volume Control System (CVCS)

- Nuclear Sampling System (NSS)
- Steam Generator Secondary Sampling System (RES)
- Safety Injection System (SIS)
- Nuclear Vent Drain System (NVDS)
- Component Cooling Water System (CCWS)
- Demineralised Water System (SED)

63 The analysis details the potential failures that could occur in the systems identified, however, a number of the systems have little or no impact on flooding within the Reactor Building due to either the limited volumes of water associated with the system or due to the bounding nature of other faults that are captured by the systems in place to protect redundant F1 safety classified systems.

64 The submission provides details of the potential pipe failures as well as the route the water takes be that either to the IRWST or stored within a retention volume to ultimately flow into the IRWST that have been subject to further detailed analysis. The "dead volumes" are not considered further given that the room can contain the water and that there are no F1 safety class systems contained therein.

65 Two rooms, [redacted], are identified where there is a need to ensure that the barriers remain water tight. Although each of the rooms do not contain F1 safety class systems, a leak within the pipe [redacted] (in the case of [redacted]) would result in draining of the IRWST into the room by gravity until such time as the level equalises. As part of the bounding analysis it is assumed that the room affected completely floods and the water then propagates into the Corium Spreading Area, [redacted]. The PCSR states the design criteria for the Corium Spreading Area is that it must be ensured that the room remains free of water prior to any potential corium spreading event. As a result of this bounding scenario, the following requirements are identified associated with ensuring that water does not propagate to the Corium Spreading Area:

- The sleeves in walls, [redacted], which run the cables connected to the valves [redacted] respectively, are required to be watertight.
- The floors of rooms, [redacted] are required to be watertight to prevent water propagated into the Corium Spreading Area.
- The drip pans, [redacted] are required to be watertight to ensure that water does not overflow to the discharge channel through a number of pipes.

66 In addition to the approach taken to the rooms that could contain water, a number of areas are analysed to determine the impact of flooding. The analysis confirms that the potential effects of flooding are acceptable either due to them containing no F1 safety class systems or that the F1 systems contained within the areas are above the maximum potential flood height when taking into account flow rates through openings within the area.

67 The engineered flow routes to the IRWST are designed such that any water released as a result of flooding would flow through openings in the structure and ensure that the water is directed to the IRWST. For these rooms only the largest initiator is considered and consequences are evaluated. The water depth in the IRWST are analysed as well as the effects of overflow at the elevation [redacted]. There are diagrams contained within the submission which identify the various routes to the IRWST from rooms in which flooding events could occur.

68 The maximum volume of water that could be released into the IRWST from an internal flooding event is [redacted] as a result of leak of the SED within room [redacted]. This corresponds to a water level at an elevation of [redacted] within Containment. The analysis identifies which rooms could be flooded as a result of the IRWST overflowing by [redacted]. It then considers what F1 safety class systems are located within the rooms in question and concludes that due to the classification of plant i.e. tolerable to flooding, or the height at which the system is located being above the maximum flood height, no F1 safety class systems are lost.

3.1.5.2 Flooding of the Reactor Building Annulus (HRB)

69 Flooding of the Reactor Building Annulus (HRB) is considered in detail within the analysis. It states that the only the following systems are potential flooding initiators within the annulus:

- Nuclear Island Fire Protection System (JPI)
- Demineralised Water System (SED)

70 The basis for the other water systems within the Annulus not being considered further is due to one or more of the following factors:

- The systems are double walled high energy lines as they pass through the annulus, or
- they do not contribute significantly to flooding of the area due to the bounding nature of the other systems analysed, or
- they are used outside of normal operation, or
- there are preventative operator actions in place to prevent water levels in the annulus affecting more than one F1 safety class system.

71 A guillotine break of a 50mm line of the Nuclear Island Fire Protection System (JPI) within the annulus is postulated for the analysis. The basis for this selection is based upon the pipe quality such that only leaks are postulated in the JPI pipework for Nominal Diameters (ND) > 50mm and hence it is considered to be bounding. The JPI pipework system enters the annulus from SABs 3 and 4, and the FB and the flow from a breached 50mm pipe is calculated to be [redacted]. In the event of such a failure there is a pressure drop in the fire hydrant line (JAC) that serves the JPI, and once the pressure detected is less than 10bar, it is detected by sensor [redacted]. At this point, alarm [redacted] is raised within the MCR and one JAC pump automatically starts. Should the pressure continue to fall to below 9 bar, sensor [redacted] triggers a further alarm, [redacted] which results in the operation of a second pump. The pressure sensors are classified as F2.

72 The submission claims that operation of the JAC pump identifies a leak in the fire protection system with sump [redacted] within the annulus identifying the location.

At this point an operator would be dispatched to investigate and isolate the following manual valves:

- Outside containment: [redacted], and
- inside containment: [redacted].
  These valves are located near to the access point [redacted] above the bottom of the annulus and it is claimed that they remain accessible in the event of flooding.

73 The claimed operator time for manual isolation achieved by local operation is 1 hour and as a result it is claimed that the amount of water that would have been released into the annulus is [redacted].

74 In the event of a leak in the demineralised water system (SED), there are no means to detect that it is from this system given that it is a non-safety classified system and that the sump within the annulus would only provide information of the presence of water. The only means, therefore, to ensure that flooding within the annulus is prevented from affecting F1 safety class systems in this case is on detection in the sump [redacted], which indicates that a preventative isolation of the system is required. The method by which this is achieved is through the closure of an F1A motorised valve and should this fail manual isolation through closure of [redacted] within the FB. The manual valve is classified F2. The resultant volume based upon the operator intervention is [redacted].

75 Furthermore it is concluded that the bounding scenario for the Reactor Building Annulus is a 50mm guillotine failure of a JPI line resulting in a water volume of [redacted] with a resulting flood depth within the annulus of [redacted]. As was the case with the Containment, it is claimed that no F1 safety class equipment is lost as a result of this flooding event providing the identified operator actions, both from the MCR and locally, are undertaken.

76 A number of assumptions are cited which form the analysis:

- The integrity of non-electrical equipment is not compromised by the flooding.
- Underwater operation of passive-operation equipment such as swing and piston check valves is assumed.
- Spraying by the JPI sprinklers does not affect operability of class F1 ducts [redacted].
- The impact of cable trays and electrical equipment connections has not been taken into account.

3.2 EDF and AREVA Submission Relating to Internal Fire Verification and Validation, GI-UKEPR-IH-02.A2

77 In response to the GDA Issue Action, **GI-UKEPR-IH-02.A2**, EDF and AREVA submitted the report, *"Location of Cable Trays in the EPR FA3 Nuclear Island"* (Ref. 20), the overview of which is detailed within this section.

3.2.1 Location of Cable Trays in the EPR FA3 Nuclear Island, EYRT2010FR0042, Revision B

78 The purpose of the above submission (Ref. 20) is to examine the Flamanville 3 EPR design in relation to the potential for cables from a foreign division being routed through one of the other divisional segregation areas. The approach taken has to use the Plant Design and Modelling System (PDMS) to pinpoint areas where such conditions exist and

to provide fire protection measures to mitigate the risk as a result of fire. The fire protection measures cited within the submission are:

- Fire load suppression wrapping to essentially remove the fire load.
- Functional cable wrapping to ensure cable operability during a fire.

79 The cable trays subject to the analysis are contained within the Reactor Building (RB), Fuel Building (FB), and the SAB. The submission confirms that there are no cables of a foreign division within the respective Diesel Buildings (DB).

80 Each of the cable trays have been subject to analysis through utilising the cable tray identifier within the PDMS as a means to determine which division the cable tray belongs and applying that to the room in which the cable tray is routed. This then results in cables trays from a foreign division being identified. There were three fire compartments identified that did not have their divisions identified within the model, namely:

- [redacted] It contains cables from Divisions 2 and 4. This has now been allocated to Division 2 and as a result the Division 4 electrical cables are considered as cables from a foreign division.
- [redacted] This non-contained area which includes reactor building pools cannot be allocated to a single division and cables from all four electrical divisions are contained within this area.
- [redacted] This fire cell includes the entire service floor and annular area on level [redacted].

81 The extent of cable wrapping to be installed within each building of the FA3 EPR design is included within Table 1.

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

Table 1: Cable wrapping to be installed within each building of the FA3 EPR design

82 The third column of the above table identifies the additional cable trays that are required to be wrapped but have not yet been identified within the PDMS model for FA3. There is conservatism in the approach applied, as the extent of wrapping undertaken assumes the entirety of the length of the cable tray is to be wrapped rather than at specific locations.

83 There is a detailed table which documents each cable tray that contains safety related cabling and identifies what division the cable tray belongs to as well as what division it passes through and any resulting requirement for cable wrapping to be provided.

84 The submission recognises that further analysis is required for the SAB Division 4 building given that not all the information in relation to cable tray routing had been entered on to PDMS at the time of producing the report, however, it recognises the need for this to be done at a later date.

3.3 EDF and AREVA Submissions Relating to High Energy Line Break Verification and Validation, GI-UKEPR-IH-02.A3

85 In response to the GDA Issue Action, **GI-UKEPR-IH-02.A3**, EDF and AREVA submitted three deliverables (Refs. 21, 22, 23) and an overview is provided of each within this section.

3.3.1 Application of the MTE175, Breaches of High Energy Pipes, load caused on the structural steelworks, ECEIG091393, Revision A1

86 The purpose of the above submission (Ref. 21) in response to GDA Issue Action 3 is to provide the methodology and rules applied to the design of structures in relation to the High Energy Line Break (HELB). It provides assumptions as well as jet effect calculations associated with hydraulic forces for pipework having, in normal operation, pressures in excess of 20 bar or temperatures above 100^{o}C, the criterion for the designation of high energy. The calculations detailed within the submission enable further analysis of potential impact on structural steelwork.

3.3.2 Additional information for the stage 1 analysis report on the consequences of high energy line breaks – fuel building, ECEF101595, Revision A1

87 The above submission (Ref. 22) provides the outcome of the studies undertaken to complete the first stage HELB studies for the Fuel Building. The additional information summarised within the report aims to:

- Finalise the studies undertaken during the first stage, with reference to the effects of jets and whipping on equipment and lines through the application of more precise assumptions in relation to the open points raised and by integrating the effect of jets caused by any longitudinal breaks.
- Taking into account the overall effects of the spread of degraded ambient conditions including the effect on equipment as well as the resistance of civil structures at given temperature and pressure differentials.

88 The analysis does not include the effects of HELB on cable routes. This is identified as needing to be undertaken when the information is available and incorporated into the second stage HELB studies.

89 The open points raised as a result of the first stage studies were:

- Risk of damage to the EBS in Division 1 by the CVCS in Division 4.
- Risk of initiating a PCC-3 event of emptying or loss of FPCS cooling after a CVCS break.
- Risk of a break in the service air pipe (SAT) affecting the ventilation of F1 functions.
- Risk of initiating a PCC-3 event of emptying or loss of FPCS cooling after a SAT break.

90 The submission cites Reference 28, *"Treatment of open points from the 1st phase HELB study in the Fuel Building"* as the source of the detailed analysis presented. For these cases it is concluded that the events would not occur due to geographical location, the presence of obstacles, the orientation of the pipe in question or the lack of adequate leverage to enable pipe whip to occur. The submission states that the second event has yet to be fully resolved due to the risk to the FPCS within rooms [redacted] and [redacted] not yet being resolved as an analysis of stresses for the pipe, [redacted] [the injection pipe to the reactor pump seals] needs to be undertaken.

91 The submission concludes, with the exception of the above outstanding analysis, that the risk is currently acceptable associated with the potential for a HELB to result in emptying or loss of FPCS after a CVCS break due to the following arguments:

- There is significant margin in the potential emptying rates given that the PCC-3 event requires make up of [redacted] tonnes of water per hour and the worst case leakage arising from pipe break of the FPCS in this event is approximately [redacted] tonnes per hour. As a result there is a long period of time before cooling of the fuel pool could be affected even if no action were to be taken to isolate the affected system.
- For the other events involving HELB, the PCC analysis detailed within Chapter 14 of the PCSR (Ref. 11) identifies that there would be no loss of cooling of the fuel pool, even on a transient basis.
- None of the safety functions used to manage the PCC are affected during the event and as a result the reactor can be brought to a controlled state and then on to a safe state.

92 To complete the first part of the analysis, the submission confirms that the FB does not contain any longitudinal welds in the high energy pipework present.

93 The potential for degraded ambient conditions within the FB has also been considered within the submission which presents the requirements in place for degraded ambient conditions to spread from:

- One safety-class building to another.
- Nuclear Auxiliary Building to a safety-class building.
- One safety-class division to another.

94 The submission concludes that the above requirements relating to prevention of the spread of degraded ambient conditions are met for the FB for the following reasons:

- Absence of spreading from a SAB to the FB, due to the preferential route designed in SABs 1 and 4 to moderate pressure and temperature conditions in rooms containing systems at over 100°C (water and steam valve bunkers, MFWS line corridor, room containing the SGBS valves and the SIS rooms);
- Absence of spreading from the NAB to the fuel building, mainly due to a preferential expansion of degraded ambient conditions in the NAB, promoted by the free volume and reinforced by the air seals on doors at the interface;
- Absence of spreading of degraded ambient conditions between the two divisions of the fuel building, due to the absence of high energy line systems potentially causing degraded conditions in the fuel building.

95 The submission provides results of overall studies undertaken, namely:

- The possible loss of several redundancy factors for one single F1 function after a HELB in the fuel building is excluded;
- Most possible PCC-3 initiation scenarios after a HELB in the FB have been excluded. Outstanding cases would lead to PCCs exclusively requiring the use of automatic and passive systems and devices to achieve controlled state and safe state. The event fails to lead to the loss of the cooling of the fuel pool, even on a transient basis. The risk is considered to be acceptable without modification;

- No situation exists, after a HELB in the fuel building, where a 7000 ppm boron injection is not possible.
- The FB does not contain any high energy lines with longitudinal welds. A possible longitudinal break is not therefore considered for this building;
- There is no risk of degraded ambient conditions spreading from the FB to another building, between two divisions of the FB, and from another building to the FB.
- No degraded conditions qualification requirement exists after a HELB in the FB because no high energy system exists where a break could lead to degraded ambient conditions;
- For the same reason, no effects on the structures relating to a change in temperature / pressure in the rooms of the FB are required to be studied for HELBs.

96 The submission does identify further analysis associated with the following two areas:

- The consideration of cables likely to be damaged by a HELB. This analysis requires the consideration of cable routes, which are not available at this stage of the project;
- The consideration of flooding induced by line breaks and leaks after a HELB. This analysis requires the consideration of flooding studies.

3.3.3 Additional information for the stage 1 analysis report on the consequences of high energy line breaks – safeguard auxiliary buildings, ECEF101596, Revision A1

97 The above submission (Ref. 23) provides the outcome of the studies undertaken to complete the first stage HELB studies for the SABs. The additional information summarised within the report aims to:

- Finalise stage 1 studies, with reference to the effects of jets and whipping on equipment and lines. This is done by applying more precise assumptions on the location of the breaks to handle the outstanding points revealed in stage 1 analyses and by integrating the effect of jets caused by any longitudinal breaks;
- Contribute initial information on the robustness of the installation of cable trays in terms of whipping and jet risks;
- Take into account the overall effects associated with spreading of degraded ambient conditions and the effects of degraded ambient conditions on equipment.

98 The analysis does not include the effects of HELB on cable route, flooding caused by system breaks, and the pressure and temperature differences supported by civil engineering structures.

99 There were two open points raised as a result of the first stage studies. The first open point was associated with the threat to the ventilation of F1 functions due to a break in the CHRS, SAT, or SED high energy line. The second being the potential initiation of a PCC-3 or PCC-4 event arising from a break in the CHRS line.

100 The first open point identified that the following F1 safety functions could be impacted by a HELB (loss of two redundancies)::

- Isolation of ventilation in the SAB controlled zone Controlled Safeguard Building Ventilation System (CSBVS) blowing and extract function.

- Start-up of CSBVS iodine filtration function.
- Isolation of blowing air supply to the room opposite the personnel airlock CSBVS.

101 The second open point identified the following potential PCC scenarios arising from a HELB:

- PCC-3 - Isolatable line break in a system connected to the spent fuel pool (states A to F).
- PCC-4 - Isolatable break in the SIS operating in residual heat removal mode (≤ DN 250), inside or outside the containment (states C and D)

102 For both open points, the submission makes reference to supporting analysis, *“High Energy Pipe Break: Treatment of open points – common mode failure and safety related aspects in the Safeguard Buildings”* (Ref. 29), which provides further detailed analysis. The second open point makes reference to the first stage analysis for the SAB (Ref. 30) for the PCC-4 event detailed above.

103 For the open points considered the following conclusions are drawn, based upon the further analysis undertaken within References 29 and 30:

- The loss of two redundancy factors for the *"Isolation of the SAB controlled zone (CSBVS blowing and extraction)"* function is excluded;
- The risk of losing more than one redundancy factor for the *"Start-up of CSBVS iodine filtration"* function is excluded in SAB 1, 2 and 3 and confirmed in SAB 4. The analysis described in the sub-section *"handling of outstanding point no. 1"* demonstrates that this situation is acceptable.
- The risk of losing more than one redundancy factor for the *"Isolation of blowing air supply to the room opposite the personnel airlock (CSBVS)"* function is confirmed. The analysis described in the sub-section *"handling of outstanding point no. 1"* demonstrates that this situation is acceptable.
- The possible initiation of the PCC-3 event, *“Isolatable line break in a system connected to the spent fuel pool (states A to F)"* can be excluded as a result of a HELB.
- The possible initiation of the PCC-4 event, *“Isolatable break in the SIS operating in residual heat removal mode (≤ DN 250), inside or outside the containment (states C and D)"* can be excluded as a result of a HELB.

104 As was the case for the FB, the submission confirms that the SABs do not contain any longitudinal welds in the high energy pipework present.

105 The potential for degraded ambient conditions within the SABs has been considered within the submission and the requirements for the analysis of potential degraded ambient conditions are identical to those for the FB, detailed within Section 3.3.2 above.

106 The layout of the SABs is such that individual SAB buildings are adjacent to each other as well as to other buildings, namely, the FB, NAB, and a number of technical galleries.

107 The submission identifies that there is a risk of formation of a degraded atmosphere in terms of pressure and temperature within the following areas:

- Upper areas of SAB 1 and 4 where the MFWS pipes are located with the MSSS, and the MFWS and MSSS valve compartments, and, in SAB 1, the SGBS valves.

- Lower areas of SAB 1 and 4 where the SIS/RHRS trains 1 and 4 are located. [redacted]

108 The submission states that there are design provisions in place associated with both areas identified which consists of engineered routes to relieve over pressure as a result of a HELB to avoid any impact on civil engineering structures. These routes include the provision of rupture discs and gates to alleviate pressure from the -9m level of SABs 1 and 4 to outside. These routes ensure that there is no spread of degraded ambient conditions to other buildings, tunnels, or divisions of the SABs.

109 One modification has been identified for the controlled safeguard building ventilation system (DWL) system to guarantee the resistance of the duct sections between the safety-classified dampers containing the degraded atmosphere in the SIS bunkers of SABs 1 and 4. [redacted]

110 In the event of a HELB, a number of electrical and C&I equipment is identified as requiring qualification against the potential event. The qualification of the plant is associated with the following events:

- Steam Line Break
- Feedwater Line Break
- Break of a SIS/RHRS line in the SIS bunkers

111 The submission details the specific electrical and C&I equipment to be protected against the potential degraded ambient conditions that could occur should one of the events, above, occur.

112 The submission concludes that there is a generally satisfactory situation with regard to HELB within the SABs further to the application of a conservative simplified approach involving the assumption of loss of functionality of all items of equipment within a room. However, there is a need to complete the analysis by:

- Consideration of cables likely to be damaged by a HELB.
- Consideration of flooding induced by line breaks and leaks after a HELB.
- The analysis of civil engineering structures at given temperatures and pressure differentials.

3.4 EDF and AREVA Submissions Relating to Internal Missile Verification and Validation, GI-UKEPR-IH-02.A4

3.4.1 EPR Internal Missiles – Risk assessment report on building structure and layout. ECEIG091634, Revision B1

113 The risk assessment report on building structure and layout (Ref. 24) has been subject to assessment within the report, *"GDA Close-out for the EDF and AREVA UK EPR™ Reactor - GDA Issue **GI-UKEPR-IH-04 Revision 2** – Consequences of Missile Generation Arising from Failure of RCC-M Components"* (Ref. 25) as it formed a key reference to the GDA Issue relating to internal missile. It is therefore not intended to include the details of the submission within this assessment report, nor is it the intention

to repeat the outcome of the assessment other than to include an overview of the conclusions contained within Reference 25.

4 ONR ASSESSMENT

114 Further to the assessment work undertaken during Step 4 (Ref. 7) and the resulting GDA Issue, **GI-UKEPR-IH-02**, this assessment focuses on the evidence associated with the verification and validation of the claims and arguments made within the March 2011 Consolidated PCSR (Ref. 11). Identified deliverables intended to provide the requisite evidence were provided within the responses contained within the Resolution Plan (Ref. 8) provided by EDF and AREVA at the end of Step 4 of the GDA.

115 This assessment has been carried out in accordance with the ONR HOW2 document PI/FWD, *"Permissioning - Purpose and Scope of Permissioning"* (Ref. 1).

4.1 Scope of Assessment Undertaken

116 The scope of the assessment as been to consider the expectations detailed within the GDA Issue, **GI-UKEPR-IH-02**, and the associated GDA Issue Actions. These are detailed within Annex 3 of this report. For each of the following areas further evidence was sought:

- Internal flooding.
- Cable routing.
- High energy line break.
- Internal Missile.

117 The scope of this assessment is not to undertake further assessment of the PCSR nor is it intended to extend this assessment beyond the expectations stated within the GDA Issue Actions, however, should information be identified that has an affect on the claims made for other aspects of internal hazards such that the existing case is undermined, these have been addressed.

4.2 Assessment

118 The deliverables submitted in response to the GDA Issue have been subject to assessment individually to ensure completeness, and conclusions have been drawn on the totality of the assessment of the response to this GDA Issue in order to give an overall judgement on the adequacy of the evidence provided to support closure of the GDA Issue.

4.2.1 Internal Flooding Verification and Validation, GI-UKEPR-IH-02.A1

119 The GDA Issue Action, **GI-UKEPR-IH-02.A1** required EDF and AREVA to provide the requisite evidence in the form of the detailed Flamanville 3 verification and validation analysis and/or other supporting documentation in support of the claims and arguments presented within Chapter 13.2 of the PCSR associated with internal flooding.

120 The GDA Issue Action stated that the response should include analysis that supports the claims and arguments relating to:

- Civil structures (including surface coatings) claimed as flood barriers.
- Watertight doors and penetrations including qualification data.
- Drains and sumps claimed to prevent damage to nuclear significant SSCs.
- Calculations in place to support any claims made on potential water volumes.

121 EDF and AREVA provided a resolution plan (Ref. 8) that stated:

"EDF/AREVA will transmit documents from the Flamanville 3 EPR studies to support the Internal Flooding case. As they form the basis of the UK EPR design, the Flamanville 3 EPR studies are applicable to the UK EPR. The dedicated UK EPR internal flooding case will be provided during the NSL phase."

122 The assessment of the following submissions is included within this section of my assessment report:

- Flooding Safety Analysis for Nuclear Auxiliary Building HNX, PF2009EN0001 (Ref. 15)
- Flooding event – Analysis of the Safeguard Building, EZT2009EN0005 (Ref. 16)
- EPR FA3 Fuel Building Internal Flooding Assessment, EYRT2009FR008 (Ref. 17)
- Study of Internal Flooding Events in the Diesel Generator Building of the Flamanville 3 EPR, EYRT/2009/FR/0032 (Ref. 18)
- Analysis of Internal Flooding within the EPR FA3 Reactor Building, EYRT2009FR0076 (Ref. 19)

123 The assumptions associated with water volumes and associated isolation times to terminate possible flooding events are addressed as part of the response to **GI-UKEPR-IH-03** (Ref. 26) which specifically deals with the need for EDF and AREVA to provide an adequate internal flooding safety case. The GDA Issue has been subject to assessment within *"GDA Close-out for the EDF and AREVA UK EPR™ Reactor - GDA Issue **GI-UKEPR-IH-03 Revision 2** – Internal Flooding Safety Case"* (Ref. 27) and as a result the focus of the assessment undertaken within this report has been associated with the methodology applied to the analysis of the flooding events rather than the specific volumes of water arising from the flooding event.

4.2.1.1 Flooding Safety Analysis for Nuclear Auxiliary Building (NAB), PF2009EN0001, Revision D

124 The focus of the assessment of the above submission (Ref. 15) is to confirm the validity of the claims that an internal flooding event cannot result in a threat to adjacent buildings containing safety significant SSCs in advance of achieving a safe shutdown state. The focus of the assessment is primarily on the passive barriers by which the water containment function should be achieved.

125 The NAB is adjacent to both SAB 4 and the FB. There are no access ways between SAB 4 and the NAB beneath 0.00m level and as a result no claims made for watertight doors beneath this level. Likewise there are no access ways between the FB and the NAB beneath 0.00m and as such no claims are required to be made for watertight doors between these two buildings. The response to **TQ-EPR-679** (Ref. 13) provided during Step 4 also confirmed that the barriers between these two buildings and the NAB provided protection against flooding to the 0.00m level and as such all penetrations would be capable of withstanding a [redacted] high water column.

126 I am satisfied that an internal flooding event within the NAB would not be able to threaten either SAB 4 or the Fuel Building and hence any safety significant SSCs. No further assessment of the NAB is, therefore, required with regard to internal flooding as part of close out of this GDA Issue.

4.2.1.2 Flooding event – Analysis of the Safeguard Building, EZT2009EN0005, Revision E

127 The above submission (Ref. 16) states that the approach taken to the analysis requires the identification of the rooms in which there is the potential for loss of more than one F1 redundant function due to flooding. Following on from this the drainage paths for the water including under doorways, through drains, HVAC or bursting discs are identified. The next stage is to consider whether the postulated flow from the failed pipe can be successfully drained away prior to affecting the F1 redundant functions; if not then the potential water column height is calculated and consequences analysed. The report provides the detailed consequence analysis together with the supporting safety justification.

128 The detailed analysis presented associated with divisional separation includes both the flood sources and respective heights within rooms. The analysis considers specific threats to divisional segregation including the assessment of flow paths, openings and overflow devices, and water-tightness of penetrations and divisional doors. In addition, where doors are located within divisional boundaries above 0.00m level, thresholds sufficient to withstand the potential water column height for the room are included as part of the civil engineering design.

129 This approach presents a clear and logical approach to the consideration of drainage paths and retention of water within each of the individual SABs to ensure divisional segregation is not compromised. Figure 1 provides an example of the drainage paths identified within SABs 2 and 3, which illustrates the drainage routes through to the lower levels from the 0.00m level.

[redacted]

Figure 1: Location of Drainage Routes within SABs 2 and 3

130 I am satisfied with the analysis and design features in place associated with drainage routes and watertight doors for the assumed volumes within the analysis.

131 At the lower levels of each of the SABs there a number of leak detection methods employed within the EPR™ design. There is leak detection provided within both the controlled and non-controlled areas. The controlled area building sump sensors are classified as F2 and are located within [redacted] sumps within [redacted] and [redacted] sumps within [redacted]. The building sump level monitoring systems have redundant sensors and alarm back to the MCR as "Sump level very high" and "Sump overloading".

132 In addition to the building sumps, there are pressure relief ducts from the SIS/RHRS rooms in each of the four SABs. These ducts perform the function of depressurising the rooms in the event of a pipe break. There are [redacted] measurement sumps together with [redacted] pressure sensors contained within each of the SIS/RHRS rooms. The automatic trip of the pumps on the SIS/RHRS is undertaken through the application of a [redacted] voting logic for both pressure and sump level measurement.

133 Within SABs 1 and 4 there are [redacted] measurement sumps installed for the Containment Heat Removal System (CHRS) which are arranged to detect leakage in the long term after a severe accident.

134 Figure 2, below, shows the location of the sumps within the -9.0m level of SAB 4

[redacted]

Figure 2: Location of Sumps within SAB 4

135 The analysis provides further information relating to the claims on barriers and doors to prevent water passage. With the exception of the two watertight doors that are claimed to withstand a [redacted] head of water, the doorways between the divisions and the other buildings are claimed to be watertight to a height of [redacted] and have the function of ensuring that water is directed either through RPE gullies or down stairwells into the

basement areas of the SABs. The barriers perform the same function in ensuring that any internal flooding event does not propagate to a neighbouring division and is directed to the basement areas of the building in which the internal flood is initiated.

136 I am satisfied that the provisions in place for leak detection utilising a number of sumps and redundant level monitoring ensures that a single flooding event within each division will be detected and not result in loss of more than one division for the water volumes assumed within the analysis.

137 The common mode failure analysis undertaken for SABs 2 and 3 identifies two rooms [redacted] in which a pipe rupture of the fire hydrant system (JPI) would result in complete flood up of the room in less than one hour and would result in the potential loss of [redacted] in both rooms. In order to prevent flooding affecting both dampers, a modification to the design is to be undertaken associated with engineering additional openings to drain water into the corridor and then down through SAB 2 or 3 depending upon the pipework failure. The two modifications that are to be undertaken for each of the rooms are:

- The seal under the fire doors should be installed to allow water flow out of rooms [redacted] ([redacted] evacuated under a potential water column of [redacted]).
- Discharge valves are to be installed to provide sufficient evacuation capacity in the event of JPI pipe break ([redacted])

138 Figure 3, shows the area and modifications to be applied.

[redacted]

Figure 3: Flow Paths arising from Failure of the JPI in either [redacted]

139 I am satisfied that the proposed modification will ensure that an internal flooding event within either of the rooms [redacted] will not result in loss of the [redacted] in both rooms. However, as this has been identified as a modification to the design, I have raised the following assessment finding to ensure that this modification is adequately captured within the UK EPR™ design:

> ***AF-UKEPR-IH-10:*** *The Licensee shall ensure that the design changes arising from the Flamanville 3 Verification and Validation process are appropriately considered within the site specific design.*
>
> ***Required Timescale:*** *"Mechanical, Electrical, and C&I Safety Systems – Before inactive commissioning".*

140 A further area that has been subject to analysis is the DFL (smoke extract and pressurisation) concrete duct that passes through each of the SABs in the corridor at the [redacted] level. The concrete duct forms the ceiling of the corridors at [redacted] level and serves to pressurise the staircases and as a result it is above the maximum flood height for each of the SABs. There is also confirmation that the fire hydrant system (JPI) is a dry system above the [redacted] level and as such water entering the duct as a result of a failure of the JPI system has been discounted.

141 The final area which has been subject to assessment from the analysis is the identified modification associated with the Nuclear Island Vent and Draining System (RPE) within SABs 2 and 3. The basis of the deterministic case for flooding is that there should be no route for water to pass between SABs such that divisional segregation could be compromised. The analysis identified one area where the drainage from SAB 2 passed through into a gulley into a SAB 3 RPE drainage system at the [redacted] level, and hence divisional segregation is compromised. The modification is therefore to disconnect the drainage route from SAB 2 to SAB 3 and to connect the drainage route to the RPE drainage path in SAB 2. In addition, the analysis confirms that all other drainage routes from SABs 2 and 3 have been confirmed as being correct and contained within their respective division.

142 It is positive to note that the thorough and robust analysis identified this safety significant shortfall and provides me with further confidence in the approach taken to the internal flooding analysis of the SABs.

143 I am satisfied that the approach undertaken for the SABs demonstrates a robust approach to the analysis of potential water flows arsing from internal flooding events within the Safeguard Auxiliary Buildings.

4.2.1.3 EPR FA3 Fuel Building Internal Flooding Assessment, EYRT2009FR0008 Revision C

144 The analysis undertaken within the above submission (Ref. 17) identifies a number of claims associated with the civil structures, most notably the ability of the external barriers and the divisional separation between the two halves of the Fuel Building being rated to withstand a [redacted] high water column arising from internal flooding events. In addition the doors installed within the divisional segregation barriers at the [redacted] level are rated to withstand the passage of flood water between the two divisions of the Fuel Building. The claims made on the doors are that they should be able to withstand the maximum flood height of the rooms on either side.

145 There is one specific area identified that requires modifications to be made to ensure that the loss of more than one F1 redundancy is prevented. This is at the [redacted] level and is associated with the removable floor slabs for pumps [redacted] and

[redacted] within room [redacted] the layout of which is shown within Figure 4.

[redacted]

Figure 4: Location of Concrete Slabs within Room [redacted]

146 The modifications identified within the submission associated with the need to ensure divisional segregation within this room are:

- Making the seals around the removable concrete slabs watertight (dotted lines in Figure 4 above).
- Avoidance of the removal of both slabs simultaneously.
- All walls and doors to the room to be watertight to [redacted] and are kept closed when a slab has been removed.
- The installation of a valve to isolate the floor drain [redacted] which is normally open and only closed when the slab to access pump [redacted] is removed.

147 Figure 5 shows a screen shot from the 3D model which illustrates the slabs, the floor drain and the two divisional pumps located beneath the [redacted] level.

[redacted]

Figure 5: 3D Model image of Watertight Floor Slabs and Floor Drain within Room [redacted]

148 Whilst I am satisfied with this approach to ensuring that there is no loss of more than one F1 redundancy within the Fuel Building, there is a need to ensure that this significant modification is captured during the site specific phase and as such assessment finding, **(AF-UKEPR-IH-10)**, again applies.

149 I am satisfied with the overall approach that has been taken to the analysis of potential internal flooding scenarios undertaken within the verification and validation studies for FA3 for the Fuel Building. The approach taken to the specific analysis of flood heights within rooms is the same as that applied for the SABs and is consistent with my expectations.

4.2.1.4 Study of Internal Flooding Events in the Diesel Generator Building of the Flamanville 3 EPR, EYRT2009FR0032, Revision E

150 Once again, the approach adopted to the verification and validation for the Diesel Generator Building (Ref. 18) is the same as that applied to the both the Safeguard Auxiliary Buildings and Fuel Building.

151 Figure 6 illustrates a section through a diesel generator building and the red lines depict the divisional segregation that is in place. The segregation essentially separates the Diesel Generator Building into three separate buildings with no penetrations passing between them. The separate compartments house each of the divisional diesel generators, and the SBO diesel.

[redacted]

Figure 6: Section through a Diesel Generator Building

152 A comprehensive room by room analysis has been undertaken as part of the verification and validation for FA3 which has considered the maximum potential flood volumes and protection of specific penetrations to ensure that no more than one F1 redundancy is lost. The internal flooding case for the Diesel Generator Buildings is more straightforward than for the other NI buildings as there is full segregation provided throughout the height of the building coupled with a further two trains contained within the other diesel generator building which is not assumed to be lost due to the same internal flooding initiator.

153 There is one area within an adjoining Technical Gallery which the analysis identifies does not meet the expectation that a flooding event from an adjacent building should not be able to propagate into a safety classified building. The submission cites that it may not be possible to achieve a watertight seal across the door at the [redacted] level due to the door being identified as a “panic door”. The submission addresses the potential for flood water originating within the Diesel Generator Buildings flooding into the technical galleries, however, it does not discuss the potential for flood water arising from the technical galleries flooding into the Diesel Generator Buildings. It is recognised that flooding arising from the Technical Galleries is out of scope of GDA, however, such an event could lead to a challenge to safety classified plant. As a result, I would expect that

this shortfall be subject to further analysis during the site specific phase to ascertain what further options could be taken forward to prevent flooding within the technical galleries affecting a division of diesel generator supplies. An assessment finding has been raised in relation to this shortfall:

> ***AF-UKEPR-IH-11:*** *The Licensee shall ensure that the further analysis of the options to prevent the spread of flood water from the adjacent technical galleries resulting in loss of safety classified plant and equipment within the Diesel Generator Buildings are captured as part of the site specific design.*
>
> ***Required Timescale:*** *"Mechanical, Electrical, and C&I Safety Systems – Before inactive commissioning".*

154 I am satisfied that the detailed approach to the analysis of potential flooding events within the Diesel Generator Buildings demonstrates that there is no potential for loss of more than one F1 redundancy as a result of an internal flooding event for the water volumes cited within the verification and validation studies. The potential for loss of safety classified equipment as a result of an internal flood within the technical galleries should be subject to further analysis and is captured as such within the above assessment finding.

4.2.1.5 Analysis of Internal Flooding within the EPR FA3 Reactor Building, EYRT2009FR0076, Revision D

155 The above submission (Ref. 19) considers each potential flood initiator within both the Reactor Building Containment (HRA) and the Reactor Building Annulus (HRA). It considers the potential flood levels arising from postulated breaches and includes consideration of the associated operator actions to terminate potential leaks. As mentioned previously within this report, there is a GDA Issue, **GI-UKEPR-IH-03**, which is considering the wider internal flooding safety case including potential water volumes and operator actions and as a result they are not taken into consideration within this assessment. This assessment focuses on the process applied in the verification and validation of potential internal flooding initiators rather than the safety case itself. As there is limited physical segregation within the Reactor Building, both HRA and HRB, there is greater reliance on the detection and isolation times. There are, however, far greater volumes in which water can be retained, which in many cases, results in far more extended timescales in which operator actions would be achievable, especially within HRA.

156 As was the case for verification and validation studies undertaken on other NI buildings, the approach taken is to analyse the potential internal flooding initiators and the associated water volumes released as a result. The studies then perform an assessment of each of the individual rooms and areas to ascertain the impact on nuclear safety. This approach results in a comprehensive analysis of the potential impact on safety classified plant and equipment, with the overriding principle being that any flooding event should not result in loss of more than one F1 redundancy. I am satisfied with the approach taken to analyse the potential flooding scenarios within the Reactor Building for the flooding initiators identified. As mentioned above, given the open nature of both buildings (HRA and HRB) the results are far more sensitive to assumptions made in relation to the amount of water that is released. As the assumptions relating to pipe break have been challenged by ONR as part of **GI-UKEPR-IH-03**, the application of this approach with more bounding flood volumes may result in the need for changes to the design to accommodate the increased flood volumes.

4.2.1.6 Conclusions

157 The GDA Issue Action required EDF and AREVA to provide the requisite evidence in the form of the detailed verification and validation analysis and/or other supporting documentation in support of the claims and arguments presented within Chapter 13.2 of the PCSR associated with internal flooding. The issue action stated:

"*The response should include analysis that supports the claims and arguments relating to:*

- *Civil structures (including surface coatings) claimed as flood barriers.*
- *Watertight doors and penetrations including qualification data.*
- *Drains and sumps claimed to prevent damage to nuclear significant SSCs.*
- *Calculations in place to support any claims made on potential water volumes.*
- *Any further defence in depth and ALARP measures that could be implemented into the design.*
- *Any identified design changes and their implementation within the PCSR."*

158 The deliverables (Refs. 15 – 19) provided addresses the need for the civil structures at the interface of each of the divisional segregated buildings as well as in other areas such as the corium spreading areas to be water-tight. This includes reference to the need to provide water tight seals, doors, barriers and cable penetrations to be adequately rated to withstand the effects of water. It also specifies the height of the barriers and doors that are to be claimed against the effects of internal flooding in terms of the need to qualify the barriers against the calculated head of water to which the barrier could be subject.

159 Consideration is given to the drainage of the buildings with detailed analysis of the discharge routes from the upper levels of buildings down into the retention volumes located in the basement areas. Within the basement areas, redundant sumps, together with safety classified level monitoring, are provided to ensure that any initiating flood event is detected and the information relayed to the MCR.

160 As part of the analyses undertaken on flooding sources, their discharge routes, and their segregation utilising water tight barriers, EDF and AREVA have calculated the potential flood volumes utilising a series of rules to ensure that they are conservative. One exception to this was identified in the approach taken to the assumption of leak rather than break of classified moderate energy pipework with a nominal diameter greater than 50mm, which has been raised through the GDA Issue, **GI-UKEPR-IH-03**, as it is concerned with the basis of the internal flooding case rather than the verification and validation of the existing case for Flamanville 3 (FA3).

161 Design changes have been identified within the studies that have been undertaken for FA3 and the need to capture these during the site specific phase have been captured through an assessment finding **(AF-UKEPR-IH-10)**. Further design changes have been identified as a result of the work undertaken by EDF and AREVA to address **GI-UKEPR-IH-03**, which are detailed within the assessment report addressing that GDA Issue.

162 I am content that the PCSR has captured the need for a detailed vulnerability analysis as part of the verification and validation.

163 To conclude, whilst I am content with the approach taken to the verification and validation studies within the area of internal flooding in that they are both thorough and comprehensive, the assumptions that have been made in relation to potential flood volumes and associated operator responses are yet to be resolved from an ONR perspective. This aspect of assessment will be undertaken within the GDA Close Out

Assessment Report for **GI-UKEPR-IH-03** (Ref. 27) relating to the internal flooding safety case.

4.2.2 Cable Routing Verification and Validation, GI-UKEPR-IH-02.A2

164 The GDA Issue Action, **GI-UKEPR-IH-02.A2** required EDF and AREVA to provide the requisite evidence in the form of the detailed Flamanville 3 verification and validation analysis and/or other supporting documentation in support of the claims and arguments presented within Chapter 13.2 of the PCSR associated with the routing of electrical cables within the EPR™ design in order to prevent a single fire resulting in loss of more than one divisional separation group.

165 The GDA Issue Action stated that the response should include analysis that supports the claims and arguments relating to:

- The routing and identification of protected cable trays.
- Justification of claims and arguments made relating to geographical separation.
- The provision of passive protection applied to cables and cable trays specifically.

166 EDF and AREVA provided a resolution plan (Ref. 8) that stated:

"To support the fire hazard case, EDF/AREVA will transmit a document from the Flamanville 3 EPR studies that will detail the routing of cable trays, identify foreign division cable trays and give details on cable trays protected against fire. As they form the basis of the UK EPR design, the Flamanville 3 EPR studies are applicable to the UK EPR. The dedicated UK EPR fire hazard case will be provided during the NSL phase."

167 One submission (Ref. 20) was provided in response to this GDA Issue Action, which has been subject to assessment within this section of my Assessment Report.

4.2.2.1 Location of Cable Trays in the EPR FA3 Nuclear Island, EYRT2010FR0042, Revision B

168 The above reference (Ref. 20) details the approach to ensuring segregation of the safety classified cables within FA3 and identifies the areas where foreign cables (cables from a different division) are routed through divisions within the nuclear island. Where such cables are identified, the submission identifies the most appropriate method of protection be that cable wrapping to remove the potential fire load or wrapping to ensure the functionality of the cables contained therein. The method of identifying the cable routes and the potential foreign cables involves the identification of cable trays utilising the Plant Design and Modelling Software (PDMS) for FA3. Whilst this method does not identify individual cables, it does identify the cable trays in which the cables for a particular division are routed. This is a good approach to adopt given the hundreds of miles of cables installed within the FA3 design, however, it is reliant on the specific cable being routed and identified correctly in the first instance. There is, therefore, a great deal of reliance put on cable pulling procedures to ensure that they are routed within the correct cable trays and this is an aspect that would have to be robust and managed effectively through the construction phase.

169 I don't believe that there is anything further that can be done within GDA to ensure that the correct cables are located on the correct trays; however, there is a need to consider the identification of safety classified cables from a divisional perspective to ensure clarity during construction and operation. One possible means by which this could be achieved would be through the colour coding of the safety classified divisional cables, thus ensuring that the correct cables were routed onto the correct cable trays and would serve to

provide a clear means by which the cables could be inspected visually to assist in commissioning and operation. This approach has been successfully applied within the existing PWR within the United Kingdom. The following assessment finding has, therefore, been raised:

> ***AF-UKEPR-IH-12:*** *The Licensee shall provide a means by which to physically identify individual safety classified cables from different safety divisions within UKEPR™ to ensure that there is visual identification of the different safety class cables for each division on plant.*
>
> ***Required Timescale:*** *"Mechanical, Electrical, and C&I Safety Systems – Before inactive commissioning".*

170 The submission identifies the following areas where cables are routed through foreign divisions due to the areas not being defined from a divisional perspective:

- [redacted]
- [redacted]
- [redacted]

171 I have elected to assess the Access Fire Compartment (SFA), [redacted], to determine whether the cable trays identified have been captured within the verification and validation work that has been undertaken. [redacted] is located between [redacted]. Figure 7 illustrates the location of this compartment.

[redacted]

Figure 7: [redacted] – Location of Foreign Division Cables

172 As a result of cables passing through this area from two divisions (electrical Divisions 2 and 3), the cable routing studies elected to allocate the SFA to Division 2. Hence all the cables that pass through this area from Division 3 should be protected. The tables included within the cable routing analysis identifies the cable trays associated with this area as being protected in the case of Division 3 and unprotected for Division 2, which confirms that the need to ensure that the principle of ensuring that any cables that pass

through foreign divisions are adequately protected. Figure 8 is an extract from Appendix 1 of Reference 20, which identifies that the cable tray, [redacted], should be protected for 2 hours against the effects of fire; and the requirement to adequately protected foreign cable trays has been captured (highlighted in green within the figure).

[redacted]

Figure 8: Extract from the Cable Tray Analysis within Appendix 1 of Reference 20

173 As can be seen within Figure 8, highlighted in red, there are areas where the requirement for cable protection is required, but has not been defined within the PDMS model. For the cable trays identified, it is recognised that they are partly protected but the protection is not sufficient to ensure that the foreign cable tray is adequately protected. Although this appears to be a shortfall, it reflects positively on the approach to the identification of cable trays passing through foreign divisions. The PDMS model identifies the clash points and recognises the extent of protection such that further protection can be applied. It should be noted that the submission has captured this shortfall only six times, as the initial design within the PDMS has largely captured the areas where clash points exist.

4.2.2.2 Conclusions

174 The GDA Issue Action required EDF and AREVA to provide the requisite evidence in the form of the detailed Flamanville 3 verification and validation analysis and/or other supporting documentation in support of the claims and arguments presented within Chapter 13.2 of the PCSR associated with the routing of electrical cables within the EPR design in order to prevent a single fire resulting in loss of more than one divisional separation group. The issue action stated:

"*The response should include analysis that supports the claims and arguments relating to:*

- *The routing and identification of protected cable trays.*
- *Justification of claims and arguments made relating to geographical separation.*
- *The provision of passive protection applied to cables and cable trays specifically.*

- *Any further defence in depth and ALARP measures that could be implemented into the design.*
- *Any identified design changes and their implementation within the PCSR."*

175 The deliverable (Ref. 20) provided in response to this action identified the cable trays in the FA3 nuclear island together with their allocated division to ensure that the required segregation was maintained. It identified exceptions to segregation associated with cable clash points within the design and identified measures through the provision of additional cable wrapping to ensure that loss of more than one redundant train was prevented. Within the submission there are no claims made with regard to geographical separation as all cables routed within common areas are subject to additional passive fire protection rather than make arguments claiming otherwise.

176 No further defence in depth and ALARP measures were identified given the segregated nature of the design coupled with the robust approach to the protection of cables that are located within foreign divisions. In addition, no changes were identified as being required for the PCSR for the GDA Issue Action.

177 I am satisfied that the approach to the identification of safety classified cable trays within foreign divisions is sufficient for the purposes of GDA. There is a need for individual safety classified cables from separate divisions to be visually identifiable on plant, however, this is a task that is to be taken forward into the site specific phase and has been identified as an assessment finding **(AF-UKEPR-IH-12)**.

4.2.3 High Energy Line Break Verification and Validation, GI-UKEPR-IH-02.A3

178 The GDA Issue Action **GI-UKEPR-IH-02.A3** sought the further evidence associated with the expectation detailed within Section 4.3.2 of the Step 4 Internal Hazards Assessment of the EDF and AREVA UK EPR™ (Ref. 7):

"I believe that the deterministic approach to loss of multiple equipment within areas vulnerable to HELB is an acceptable approach in the first instance as a means to identify any vulnerability associated with HELB. However, I would expect any areas identified within the 1st stage analysis to be captured and analysed at the earliest opportunity in order to ensure that the specific threats associated with HELB are designed out in the first instance."

179 The GDA Issue Action, **GI-UKEPR-IH-02.A3** required EDF and AREVA to provide the requisite evidence in the form of the detailed Flamanville 3 verification and validation analysis, specifically, the FA3 First Stage Pipe Break Analysis and/or other supporting documentation in support of the claimsand arguments presented within Chapter 13.2 of the PCSR associated with high energy line break (HELB) within the EPR design.

180 The GDA Issue Action stated that the response should include analysis that supports the claims and arguments relating to:

- Consequence analysis, where applicable.
- Break preclusion.
- Identification and qualification of physical restraints, barriers and doors.
- Identification and qualification of pressure relief panels/routes.

181 EDF and AREVA provided a resolution plan (Ref. 8) that stated:

"EDF/AREVA will transmit additional documents from the Flamanville 3 EPR project to support the High Energy Line Break case. As they form the basis of the UK EPR™

design they are applicable to the UK EPR™. The dedicated high energy line break case will be provided during the NSL Phase."

182 The following submissions were provided in response to the GDA Issue Action and consideration of each is included within this section of my assessment report:

- Application of the MTE175, Breaches of High Energy Pipes, load caused on the structural steelworks, ECEIG091393 (Ref. 21)
- Additional information for the stage 1 analysis report on the consequences of high energy line breaks – fuel building, ECEF101595 (Ref. 22)
- Additional information for the stage 1 analysis report on the consequences of high energy line breaks – safeguard auxiliary buildings, ECEF101596 (Ref. 23)

4.2.3.1 Application of the MTE175, Breaches of High Energy Pipes, load caused on the structural steelworks, ECEIG091393, Revision A1

183 The above submission (Ref. 21) provides the methodology and rules in place in relation to the second stage analysis of the HELB studies. It provides first principle hydraulic calculations for differing pipework geometries assuming differing temperatures and pressures. I have chosen not to assess this submission in any detail as the focus of my assessment and the GDA Issue Action is associated with the additional information provided in support of the first stage studies for HELB as these are more relevant to the assessment of the generic design.

4.2.3.2 Additional information for the stage 1 analysis report on the consequences of high energy line breaks – fuel building, ECEF101595, Revision A1

184 The above submission (Ref. 22) identifies the following open points raised as a result of the first stage studies:

- Risk of damage to the EBS in Division 1 by the CVCS in Division 4.
- Risk of initiating a PCC-3 event of emptying or loss of FPCS cooling after a CVCS break.
- Risk of a break in the service air pipe (SAT) affecting the ventilation of F1 functions.
- Risk of initiating a PCC-3 event of emptying or loss of FPCS cooling after a SAT break.

185 The analysis provided does not apply any claims for break preclusion of the pipe work given that the PCSR makes no claim on break preclusion other than for high integrity components associated with the reactor coolant system pipework and to the main steam lines as detailed within the Chapters 5.2 and 10.5 of the PCSR respectively. Discounting failure of HIC was subject to detailed assessment during previous Steps of GDA by structural integrity specialists.

186 The submission provides a summary of the work that has been undertaken with regard to addressing the above open points. The detailed analysis that supports the submission of the open points is included within Reference 28, *"Treatment of open points from the 1st phase HELB - Fuel Building"*.

187 A TQ, **TQ-EPR-1514** (Ref. 13), was raised requesting a copy of Reference 28 which, along with Reference 22, has been subject to assessment within this section of my Assessment Report.

188 The first open point identified is associated with the risk of damage to both trains of the Extra Borating System (EBS) by pipework associated with the Chemical and Volume Control System (CVCS). Reference 22 reviews the location of the CVCS pipework in relation to the EBS in room [redacted] and cites that it is not possible for the CVCS line within the room to result in a break in either of the EBS lines in the same location due to the configuration of the pipes and the presence of an obstacle in the form of the CVCS charging pump. The evidence demonstrating this is contained within Reference 28, the assessment of which is included below.

189 Reference 28 provides analysis of 25 potential HELBs that could result in a PCC3 event and 5 that could result in loss of more than one F1 function for the open points identified above. The report concludes that for all bar one of the potential PCC-3 events, the current design is such that the PCC3 events would not occur. For the one outstanding potential PCC-3 event arising from a HELB, there is a need for a stress analysis to be undertaken on the high energy line [redacted] to ascertain whether a break in this line would impact the FPCS within room [redacted]. The submission concludes that the potential loss of more than one F1 redundancy in the five areas identified is not possible due to the location of the high energy lines coupled with the distance and geometric considerations including the provision of barriers to impact.

190 As part of my assessment I have considered the evidence associated with the following two open points:

- Risk of damage to the EBS in Division 1 by the CVCS in Division 4.
- Risk of initiating a PCC-3 event of emptying or loss of FPCS cooling after a CVCS break.

191 The first was selected as the potential to lose the F1 function associated with Start-up of an EBS train. The second was selected as this considered the potential for a PCC-3 event associated with the FPCS and focussed on the potential HELB associated with the a break in the high energy line [redacted] due to the need to perform further stress analysis.

192 Figure 9, illustrates the location of the EBS line [redacted] in relation to the CVCS pipework [redacted] that could whip and impact the EBS line.

[redacted]

Figure 9: Analysis of hazard posed to the EBS by the CVCS in room [redacted]

193 It can be noted from the figure that a break in the high energy CVCS line [redacted] would not impact the Division 1 EBS line [redacted] due to the location of the moderate energy line [redacted] and the CVCS charging pump. The safety chilled water system (DEL) pipe supports [redacted] and [redacted], which provide a DEL feed to the EBS for Division 4, would also not be affected due to the location of the CVCS charging pump and in the event of a break of [redacted] at a single point the jet impingement would not be directed against those lines.

194 I am satisfied that there would be no loss of more than one F1 redundancy associated with pipewhip or jet impingement as a result of a failure of the high energy CVCS line identified. Further, the analysis confirms that in the event of failure of this line, neither of the two EBS lines routed within this room would be affected.

195 Figure 10 illustrates the location of the event detailed within the second open point associated with the potential break of the high energy line [redacted] and the subsequent impact on the FPCS.

[redacted]

Figure 10: Impact of CVCS High Energy Line [redacted] on the FPCS in room [redacted]

196 The two FPCS lines, [redacted] and [redacted] have been identified as being vulnerable to a break in the high energy line [redacted]. The analysis provided within Reference 28 provides details of the functional requirements of both the FPCS lines and considers the operational considerations to be taken into account. The FPCS line [redacted] is isolated by valve [redacted] during normal operational conditions and as a result there is no potential for drainage of the spent fuel pool should the line fail as a result of impact from [redacted]. The valve, [redacted], is only opened for purification of the IRWST using pump [redacted]. Should there be a need for simultaneous purification of both the IRWST and fuel pool, the valve [redacted] would remain closed as pump [redacted] would be used to perform the purification function of the IRWST. In this case a break in line [redacted] would not result in a direct path for water to be lost from the fuel pool through the breached pipe.

197 I am satisfied that the analysis in this case would not lead to the PCC-3 event associated with emptying or loss of the FPCS, given the need for valve, [redacted], to be open.

198 The high energy CVCS line, [redacted], is pipework which has a diameter of [redacted] and a wall thickness of [redacted] and the potential for it to impact on the FPCS line, [redacted] has been identified. [redacted] has a diameter of [redacted] and a wall thickness of [redacted] and as a result the analysis identifies the need for further stress analysis to be undertaken to determine if the FPCS line is at risk since the assumption that pipework with a smaller thickness cannot result in failures of those with a larger wall thickness, cannot be applied in this case. The report has then considered the consequences of failure of the FPCS line and concluded that they are acceptable.

199 I believe it to be positive that the method applied to the verification and validation of HELB uses bounding scenarios of loss of all the equipment in the room to produce a subset of areas where potential shortfalls could exist. This is a good example of a conservative approach for identifying potential threats to nuclear safety functions.

200 I have raised an assessment finding to ensure that any design changes arising from further FA3 analysis undertaken are captured within the UK EPR™ during the site specific phase:

> ***AF-UKEPR-IH-13:*** *The Licensee shall ensure that the further analyses required arising from the Flamanville 3 Verification and Validation process are appropriately considered within the site specific design.*
>
> ***Required Timescale:*** *"Mechanical, Electrical, and C&I Safety Systems – Before inactive commissioning".*

201 The analysis undertaken associated with degraded ambient conditions detailed within Reference 22 provides clear requirements for ensuring that nuclear safety requirements are not compromised through applying a straightforward logic which follows the principles of building and divisional segregation. In addition, the submission provides a detailed justification why the principles would not be threatened in the event of a high energy line break within the Fuel Building.

202 I accept that further work is required associated with the consideration of the effects of HELB on internal flooding as well as on cable routes. Given the lack of detailed information currently available for both these areas, this is understandable, however, these are aspects that I would expect to be considered during the site specific phase, hence the assessment finding, **(AF-UKEPR-IH-13)** again applies.

203 I am satisfied with the approach taken for the verification and validation of the potential for high energy line breaks within the Fuel Building of UK EPR™ as the approach is bounding in the first instance, which enables the identification of specific areas for further analysis. The majority of areas which have been subject to the further analysis have shown that the effects of a HELB are tolerable in relation to nuclear safety.

4.2.3.3 Additional information for the stage 1 analysis report on the consequences of high energy line breaks – safeguard auxiliary buildings, ECEF101596, Revision A1

204 The above submission (Ref. 23) identifies that there were two open points raised as a result of the first stage studies. The first open point was associated with the threat to the ventilation of F1 functions due to a break in the [redacted] high energy line. The second being the potential initiation of a PCC-3 or PCC-4 event arising from a break in the [redacted] line.

205 As was the case for the Fuel Building, the submission provides a summary of the work that has been undertaken with regard to addressing the above open points. The detailed analysis that supports the submission of the open points is included within Reference 29, *"High Energy Line Breaks: Treatment of the outstanding points – common mode failure and other safety related aspects in the Safeguard Buildings".*

206 A TQ, **TQ-EPR-1503** (Ref. 13), was raised requesting a copy of Reference 29, which, along with Reference 23, has been subject to assessment within this section of my Assessment Report.

207 The first open point identified three areas within the Safeguard Auxiliary Buildings associated with vulnerability of HVAC systems from HELB with the potential to threaten more than one redundant F1 system. The extent of the analysis performed on potential

HELB locations given small bore (ND<50mm), which requires consideration of a break at any point in the line, is extensive. The HVAC systems identified are associated with the Safeguard Building Ventilation System (DWL). There is detailed analysis of the potential bending moments together with supporting calculations based upon whether there is the potential for pipe whip to occur given the potential break location. These aspects are considered for multiple locations along the length of pipework with an ND<50mm. This is then followed by the consideration of the effects on the DWL dampers in the rooms identified as being at risk from loss of more than one F1 redundancy. The analysis considers the effects from both the whipping pipe and jet impingement at the postulated locations where pipe whip has been shown to be possible.

208 One example of the approach applied within the analysis is associated with a guillotine break of the [redacted] within [redacted]. The [redacted] is a non-classified high energy system and as a result pipe break has been postulated along the length of the pipework within the room. Figure 11 illustrates the location of the [redacted] whose effects of failure on the two DWL dampers [redacted] have been analysed.

[redacted]

Figure 11: Potential for Break in line [redacted] to affect [redacted]

209 As the pipework is not classified guillotine break is postulated at a number of locations. Figure 12 shows a plan layout of the area and identifies the points along the [redacted] at which break has been postulated for the purposes of the analysis.

[redacted]

Figure 12: Locations analysed for guillotine break of the [redacted]

210 Each of the locations have then been analysed to determine the impact on the DWL dampers and consider the potential for loss of F1 function. The report identifies that there would not be loss of more than one F1 redundancy for each of the potential break locations.

211 I am satisfied that the approach taken with the verification and validation for HELB affecting F1 functions within the SAB has been shown to be comprehensive with detailed calculations performed to determine the impact both in terms of pipe whip and jet impingement.

212 The second open point identified the following potential scenarios arising from a HELB within the SAB:

- PCC-3 - Isolatable line break in a system connected to the spent fuel pool (states A to F).
- PCC-4 - Isolatable break in the SIS operating in residual heat removal mode (≤ DN 250), inside or outside the containment (states C and D)

213 The PCC-3 event is associated with a HELB in CHRS system impacting on the third train of the FPCS. The third train of the FPCS is only required for maintenance purposes or when there is an RRC-A event involving either SBO, Loss of Ultimate Heat Sink (LUHS), or Total Loss of Cooling Chain (TLOCC). In line with the principles applied for the UK EPR™, an internal hazard is not considered to occur simultaneously with an RRC-A event, hence, the analysis considers HELB in the event of maintenance only. The submission states that the location of the break has been calculated where the stress intensity is highest and at the extremities of the piping.

214 Figure 13 illustrates the location of the high energy pipework within room [redacted] where the third train FPCS pump is located together with the CHRS pipework that feeds into the Reactor Building. The submission states that the only FPCS pipework that can be damaged within this room is of small diameter (≤25mm) as a result of a HELB on the

CHRS. It is not clear why this is the case, however, from the PDMS model it appears that the pipework for the CHRS Intermediate Cooling Line within this room is a smaller diameter to that of the FPCS pipework, and a HELB on the CHRS system is not expected to result in a break of the FPCS lines contained within the room.

[redacted]

Figure 13: Location of 3rd Train FPCS Pump and CHRS Pipework within Room [redacted]

215 Consideration is given to breaks in the CHRS lines within the FPCS heat exchanger room [redacted] at the point at which they enter the heat exchanger. Figure 14 illustrates the potential break locations postulated for HELB of the CHRS in this instance.

[redacted]

Figure 14: Location of HELB arising from failure of CHRS Pipework within Room [redacted]

216 The analysis identifies that the PCC event, *"Isolatable* piping *failure of a system connected to the Fuel Pond"* is possible within rooms [redacted] and [redacted], however the following mitigation is in place to ensure that should the event occur, there would be no detrimental effect on nuclear safety:

- There are two isolation valves which are physically separated located within the Fuel Building upstream of the leak or break. The first [redacted] is a manual valve and the second [redacted] is motorised. Both valves are safety classified F1B.
- Maintenance of only one FPCS train can be undertaken at power operation and the redundant train is started to ensure the continuity of the cooling process in the event of failure of the operational train.
- Loss of the third train is acceptable providing the leak can be isolated through operation of the isolation valves within the Fuel Building. There are no siphon breakers on the third train and as a result, operation of the isolation valves is essential in the event of a failure of this train.
- The leak is detected by two methods, the first being the redundant F2 level sensors within the CHRS sump located at the [redacted] level of SAB 1, and the second being the F1A level sensors located within the Spent Fuel Pool.

217 I am satisfied that the PCC-3 event arising from HELB of the CHRS would not result in a threat to the levels within the Spent Fuel Pool providing the above mitigation actions are successful. The classification and categorisation of the valves and the level sensors as part of the broader assessment finding associated with classification and categorisation being addressed within the cross cutting GDA issue, **GI-UKEPR-CC-01**.

218 The PCC-4 event, "Isolatable break in the SIS operating in residual heat removal mode (≤ DN 250), inside or outside the containment (states C and D*)"*, has been subject to assessment within this section of my assessment report.

219 The submission states that the above PCC-4 event could occur during reactor states C and D. At reactor state C temperatures of the primary circuit range from 120^{o}C to 55^{o}C and pressures from 30 bar to 1 bar respectively. At reactor state D temperatures range from 55^{o}C to 15^{o}C and the pressure is at 1 bar. The analysis states that there are two possible scenarios that could result in the PCC-4 event, namely:

- At reactor state C, a rupture in the RHR pipework thus inducing an internal hazard through the PCC-4 event itself, or
- At reactor states C and D whereby a pipe whip or jet impingement arising from failure of the CHRS Intermediate Cooling Line induces the PCC-4 event.

220 Both potential scenarios are considered within the analysis. Figure 15 illustrates the location of the pipework within room [redacted]

221 In order to ensure that the transient can be managed during reactor states C1 and C2, the following systems are required:

- Detection via the redundant SIS sump level monitoring.
- Containment and primary isolation valves as well as the RHR pump trip.
- The ability of the safety injection to be performed by the Medium Head Safety Injection (MHSI) pumps.
- At least one Low Head Safety Injection (LHSI) pump.

222 The analysis confirms that these functions would remain available as the HELB on the RHR would only result in loss of one SIS train or one of the [redacted] redundant sump level measurements.

223 In reactor state C3, there is an assumption within the PCSR (Ref. 11) that three of the SIS trains are lost due to adverse suction conditions. This assumption, together with the application of the single failure criterion for the fourth SIS train, results in a requirement for long term heat decay removal through the manual initiation of both the MHSI to ensure safety injection with pressure relief through the pressure relief valves and the two trains of CHRS to maintain the IRWST temperature [redacted].

[redacted]

Figure 15: SIS Break in Room [redacted]

224 For this postulated scenario, the analysis considers the potential for a HELB break in the RHR at locations (1) and (2) and their potential to impact the CHRS line as illustrated in Figure 15.

225 In the event of a break at location (1) the pipe would whip in an upward direction and in the case of a break at location (2) the pipe would whip in a downwards direction. As a result the potential would not exist for pipewhip on to the CHRS line, [redacted].

226 For break in any other locations it is stated that a HELB of the RHR line would result in impact of the pipe on to the structure and not onto CHRS line, [redacted].

227 I am satisfied that this assertion seems reasonable given the proximity of the piping to the walls in other locations as well as the location of the line [redacted].

228 The analysis concludes that in reactor state C3, the two CHRS trains are available to manage the PCC-4 transient.

229 I am satisfied with the approach taken to an SIS break within room [redacted] as it has considered the potential loss of the CHRS line during all potential operational states in which the PCC-4 event could be initiated. The analysis presents a thorough and comprehensive approach which has considered in detail the potential HELBs and their impact on the CHRS.

230 For the scenario involving pipewhip or jet impingement arising from failure of the CHRS intermediate cooling line at reactor states C and D inducing a PCC-4 event, the analysis considers the effects within a number of rooms within the SAB. I have elected to sample the potential impact of a HELB of the CHRS Intermediate Cooling Line on the SIS line [redacted] within room [redacted] Figure 16 illustrates the location of the CHRS Intermediate Cooling Line (in red), the CHRS main line (in yellow), and the SIS line (in pink). The CHRS Main Line is not used in normal operation, however, the CHRS Intermediate Cooling Line is a high energy line and the break is postulated at the connection with the CHRS Heat Exchanger on this system.

[redacted]

Figure 16: Threat to SIS Pipe [redacted] arising from a HELB of CHRS Intermediate Cooling Line within Room [redacted]

231 The analysis confirms that a break at the CHRS Heat Exchanger is located sufficiently far enough away such that break resulting in a pipe whip and subsequent jet impingement would not threaten the SIS line in question. The line is [redacted] away from the break location and the whipping pipe could only whip around 1.5m from its rotation point.

232 The analysis concludes that the PCC-4 event associated with a HELB affecting the SIS in this instance is not credible and that the only consequence associated with a HELB on the CHRS Intermediate Cooling Line would be loss of one train of the CHRS and loss of the third FPCS train if the event were to occur in SAB 1.

233 Again, I am satisfied with the approach taken to the analysis of this potential PCC-4 event and that the detailed layouts, coupled with the operational state of the plant, have informed the detailed assessment of the consequences of HELB.

4.2.3.4 Conclusions

234 The GDA Issue Action required EDF and AREVA to provide the requisite evidence in the form of the detailed Flamanville 3 verification and validation analysis, specifically, the FA3 1st Stage Pipe Break Analysis and/or other supporting documentation in support of the claimsand arguments presented within Chapter 13.2 of the PCSR associated with high energy line break (HELB) within the EPR design. The issue action stated:

"The response should include analysis that supports the claims and arguments relating to:

- *Consequence analysis, where applicable.*
- *Break preclusion.*
- *Identification and qualification of physical restraints, barriers and doors.*
- *Identification and qualification of pressure relief panels/routes.*
- *Any further defence in depth and ALARP measures that could be implemented into the design.*
- *Any identified design changes and their implementation within the PCSR."*

235 The deliverables (Refs. 21 – 23) provided in response to this action detail the analyses that have been undertaken associated with high energy line break. The analyses consider the consequences of failure of high energy pipework and the associated effects on adjacent pipework together with the associated PCC events that may occur as a result. Detailed analyses are presented in relation to the identification and qualification of existing structures including the location of restraints/supports and the presence of obstructions and barriers.

236 As there are no claims associated with break preclusion, there are hence no arguments in place to assess, given the assumptions associated with failure of the pipework. This does not include the HIC claims associated with the reactor coolant system pipework and to the main steam lines which have already been subject to assessment by structural integrity specialists within ONR.

237 There are pressure relief panels provided with discharge routes to outside engineered within the structure. An example of this is within SABs 1 and 4 associated with pressure relief from a pipe break in the basement associated with the Safety Injection System (SIS). In addition to this there are blow out panels for pressure relief in the event of pipe break in the Main Feedwater System (MFWS) compartments which have already been considered within the Step 4 Internal Hazards Assessment of the UK EPR™ (Ref. 7).

238 In the majority of cases, due to the segregated design no further analyses have been deemed necessary and the analyses provided in response to this GDA Issue Action have not identified further design changes. There are areas where further analyses are required and the need for these has been captured through raising an assessment finding **(AF-UKEPR-IH-13)**. In addition, no changes were identified as being required for the PCSR for the GDA Issue Action.

239 I am therefore satisfied that the approach taken in the area of high energy line break has been comprehensive and, as a result, I conclude that this GDA Issue Action has been adequately resolved.

4.2.4 Internal Missile Verification and Validation, GI-UKEPR-IH-02.A4

240 The GDA Issue Action, **GI-UKEPR-IH-02.A4** required EDF and AREVA to provide the requisite evidence in the form of the detailed Flamanville 3 verification and validation analysis and/or other supporting documentation in support of the claims and arguments presented within Chapter 13.2 of the PCSR associated with internal missiles. The issue action stated:

"The response should include analysis that supports the claims and arguments relating to:

- *Identification of all potential sources of internal missile which could result in a threat to nuclear safety significant SSCs.*
- *Consequence analysis, where applicable.*
- *Break preclusion.*
- *Identification and qualification of physical restraints, barriers and doors.*
- *Any further defence in depth and ALARP measures that could be implemented into the design.*
- *Any identified design changes and their implementation within the PCSR."*

241 The deliverable (Ref. 24) provided in response to this GDA Issue Action was subject to assessment as part of the GDA Close-Out of **GI-UKEPR-IH-04** (Ref. 25). The detailed assessment of this report can be found within Reference 25, however, the main conclusions were:

- The approach to the assessment of the quantitative consequences of the most bounding missile scenarios is in line with UK expectations and those detailed within the HSE SAPs and international guidance.
- The consequence analysis undertaken has identified the most onerous potential missile events and the calculations performed for the potential missiles have been comprehensive.
- The analysis has identified claims upon barriers made upon barriers as a means to prevent propagation of missiles and an assessment finding **(AF-UKEPR-IH-08)** has been raised requesting such barriers to be specifically identified within the safety case.
- The failure mechanisms that result in the generation of missiles are considered to be reasonable and bounding.

242 As was the case for pipe break there are no claims made in relation to break preclusion other than for the reactor coolant system pipework and to the main steam lines which have already been subject to assessment by structural integrity specialists within ONR.

243 No further defence in depth and ALARP measures were identified given the segregated nature of the design and no changes were identified as being required for the PCSR for the GDA Issue Action.

244 The report concluded, *"Further to the GDA Issue, **GI-UKEPR-IH04** and receipt of the deliverables detailed within the* Response *Plan together with the responses to the TQs raised, I am satisfied that the safety case for internal missile for the UK EPR™ is adequate. One assessment finding has been raised in relation to the identification of the barriers claimed within the analysis undertaken to prevent missiles impacting on safety related plant and equipment."*

4.3 Comparison with Standards, Guidance, and Relevant Good Practice

245 Given that the SAPs cover the full range of internal hazards, it is not intended to repeat each of the SAPs for every internal hazard that has been subject to assessment within this assessment report.

246 The significant SAPs that are applicable to all four internal hazards verification and validation areas that have been assessed within this report are discussed below. The

SAPs specific to each of the internal hazard verification and validation areas which have been subject to assessment are detailed within separate sub-sections.

247 The following SAPs have been used to inform my assessment and an analysis is provided against each in relation to the UK EPR™ design:

Engineering principles: key principles	Safety measures	EKP.5
Safety measures should be identified to deliver the required safety function(s).		

"Safety should be secured by characteristics as near as possible to the top of the list below:

a) Passive safety measures that do not rely on control systems, active safety systems or human intervention.

b) Automatically initiated active engineered safety measures.

c) Active engineered safety measures that need to be manually brought into service in response to the fault.

d) Administrative safety measures (see paragraph 376 f.).

e) Mitigation safety measures (e.g. filtration or scrubbing).

Note: The hierarchy above should not be interpreted to mean that the provision of an item towards the top of the list precludes provision of other items where they can contribute to defence in depth."

248 The PCSR (Ref. 11) together with the submissions provided in response to this GDA Issue confirm that the design hierarchy within EKP.5 has been addressed in the design of UK EPR™ given that the overriding principle is associated with the passive physical segregation of redundant safety classified plant and equipment. Where this has not been achieved detailed analyses have been provided that follow the principles of EKP.5 which demonstrate that there would be no loss of redundant safety classified equipment as a result of an internal hazard.

249 SAPs EHA.6 and EHA.14 state:

Engineering principles: external and internal hazards	Analysis	EHA.6
Analyses should take into account simultaneous effects, common cause failure, defence in depth and consequential effects.		

Engineering principles: external and internal hazards	Fire, explosion, missiles, toxic gases etc – sources of harm	EHA.14
Sources that could give rise to fire, explosion, missiles, toxic gas release, collapsing or falling loads, pipe failure effects, or internal and external flooding should be identified, specified quantitatively and their potential as a source of harm to the nuclear facility assessed.		

250 The above principles have been shown to have been considered and analysed in detail for the areas of verification and validation addressed within my assessment. The analyses have been shown to consider all the aspects of EHA.6 and subjected them to a detailed study to demonstrate that their potential source of a harm to a nuclear facility have been addressed in accordance with EHA.14.

Engineering principles: design for reliability	Redundancy, diversity and segregation	EDR.2
Redundancy, diversity and segregation should be incorporated as appropriate within the designs of structures, systems and components important to safety.		

Engineering principles: design for reliability	Single failure criterion	EDR.4
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.		

251 The UK EPR™ design, through the reference FA3 design, has shown that both these principles have been addressed from an internal hazards perspective and that the design is robust to the effects of hazards as the principles of divisional segregation and redundancy have been clearly demonstrated within the design. With regard to the application of single random failure, this too has been shown and applied within the internal hazards area to demonstrate that the design is robust to both a single random failure and a concurrent internal hazard.

4.3.1 Internal Flooding

252 NS-G-1.11 (Ref. 5) states within paragraph 3.81 and 3.82:

"All possible PIEs [Potential Initiating Events] should be carefully identified. The best approach is to base the list of PIEs on a list of SSCs and then to identify all the possible sources of liquid (water in the case of pressurized water reactors and boiling water reactors), including sources in other rooms. This identification should be supported by room by room walk-downs.

For all PIEs, unless P1 [Probability of a leak or break] is acceptably small, a liquid level as a function of time should be determined not only for the room with the source of the liquid but also for all rooms to which the liquid could spread (through doors, pipe conduits or cracks in walls or floors). In the case of breaks in pipes connected to tanks or pools, account should be taken of possible siphoning effects, which can increase the amount of liquid drained. Possible blocking of drain holes by debris should be taken into account if this would lead to more severe conditions. In determining the liquid level using a volume–height relation, the as-built status of the room should be used. The possible collection of liquid in upper parts of the room (e.g. in cable trays) should also be analysed. In some cases it may be necessary to analyse the flooding also with regard to the transport of objects and/or small particles to undesired locations. A typical example is the blockage of the strainers of the emergency core cooling system. Isolation debris, corrosion particles and even human hair can be transported by water and can block the strainers."

253 The verification and validation considered the expectations of the IAEA guidance above in considering potential sources of flooding as well as leak times, drainage routes and consideration through a room by room analysis.

254 SAPs EHA.15 states:

Engineering principles: external and internal hazards	Fire, explosion, missiles, toxic gases etc – effect of water	EHA.15
The design of the facility should prevent water from adversely affecting structures, systems and		

components important to safety.

"The design of the facility should include adequate provision for the collection and discharge of water reaching the site from any design basis external event or internal flooding hazard or, if this is not achievable, the structures, systems and components important to safety should be adequately protected against the effects of water."

255 In addition, the SAPs state within ERL.3:

Engineering principles: reliability claims	**Engineered safety features**	**ERL.3**
Where reliable and rapid protective action is required, automatically initiated engineered safety features should be provided.		

"For requirements that are less demanding or on a longer timescale, operator actions or administrative control may be acceptable to complement the engineered systems. The objective should be to minimise the dependence on human action to maintain a safe state."

256 Although the detailed internal flooding safety case for UK EPR™ has not been assessed within this report, the approach to the verification and validation for internal flooding approach is in line with my expectations. Specifically, as detailed within SAPs, specifically, EHA.6, EHA.14, and EHA.15, in relation to the analysis and sources of potential internal flooding events as well as the design of the facility to withstand the effects of postulated volumes used within the analysis. In addition, the approach conforms to the principle of ensuring a hierarchy of safety measures to ensure that passive and engineered protection measures as detailed previously within EKP.5 take precedence over administrative controls as stated within ERL.3.

4.3.2 Protection of Cable Routes

257 NS-G-1.7 (Ref. 5) considers the protection of cable routes within Appendix IV, in which it states in paragraph IV.2:

"Various design approaches have been taken to limit the significant impact of cable fires. Among these approaches are: protecting electrical circuits against overload and short circuit conditions; limiting the total inventory of combustible material in cable installations; reducing the relative combustibility of cable insulation; providing fire protection to limit fire propagation; and providing separation between cables from redundant divisions of safety systems, and between power supply cables and control cables."

258 In addition, the guidance states in paragraph IV.5:

"In some circumstances, specific passive protection measures may be necessary to protect electrical cables from fire. Such measures include:

— *Cable coatings to reduce the potential for ignition and flame propagation,*

— *Cable wraps to provide segregation from other fire loads and from other systems,*

— *Fire stops to limit flame propagation."*

259 Furthermore, EHA.17 states:

Engineering principles: external and internal hazards	**Fire, explosion, missiles, toxic gases etc – use of materials**	**EHA.17**

Non-combustible or fire-retardant and heat-resistant materials should be used throughout the facility.

260 In addition, the ONR Technical Assessment Guide, T/AST/014 (Ref. 3), on internal hazards states:

"In order that items important to safety will have the level of reliability required to meet the safety goals, the licensee must consider the possibility of single random failures, common cause failures, simultaneous and consequential events and unavailability of SSCs due to maintenance activities. Common causes include both SSC failures and effects of internal hazards such as fire. The appropriate level of reliability of essential safety functions may be achieved by incorporating redundancy within single trains and/or segregation and diversity between trains."

261 Complimenting the statements made within ONR and IAEA guidance, Western European Nuclear Regulators' Association (WENRA) Reference Level S: Protection against internal fires (Ref.4), state within its basic design principles:

- *SSCs important to safety shall be designed and located so as to minimize the frequency and the effects of fire and to maintain capability for shutdown, residual heat removal, confinement of radioactive material and monitoring of plant state during and after a fire event.*
- *Buildings that contain equipment that is important to safety shall be designed as fire resistant, subdivided into compartments that segregate such items from fire loads and segregate redundant safety systems from each other. When a fire compartment approach is not practicable, fire cells shall be used, providing a balance between passive and active means, as justified by fire hazard analysis.*

262 In addition, existing UK nuclear power generation facilities apply a similar approach to ensuring that there is adequate redundancy and segregation in place to ensure that the design basis stated above is met.

263 The approach to divisional segregation of safety classified cable trays from foreign divisions has been addressed within the FA3 design as part of the principles applied to ensure divisional segregation for all safety classified equipment. When this has not been possible on limited occasions, the approach to provide additional passive protection has been applied based upon the cable tray routing information derived from the PDMS model for the FA3 design.

264 The analyses undertaken for the protection of cable trays within FA3 has demonstrated that the expectations of the SAPs, specifically, SAP EHA.5, EHA.6, and EHA.17 have been addressed and that the expectations detailed within IAEA guidance have been met through the detailed and comprehensive analyses that have been undertaken by EDF and AREVA when considering segregation of safety classified cable routes within the FA3 design. This approach provides confidence in the approach to be taken for UK EPR™ when considering the protection of safety related cable trays within the design.

4.3.3 High Energy Line Break

265 NS-G-1.11 (Ref.5) states within paragraph 3.55:

"The whipping pipe branches should be analysed geometrically to determine possible directions of motion that might endanger target SSCs, as well as to evaluate their kinetic energy. Any possible mechanical impact on the target should be investigated by means of an appropriate dynamic analysis made on the basis of a detailed assessment of the

system transient, to quantify the discharge forces and the energy of the whipping pipe as well as the fraction of the energy that would be transferred to the target (the extent of the analysis can be limited on the basis of conservative assumptions). In addition, the analysis should include an assessment of the effectiveness of the pipe whip restraints, demonstrating that pipe deflections may be kept small by the physical restraints. In the case of terminal end breaks, consideration should be given to the secondary effects on the remaining terminal ends."

266 The IAEA guidance detailed above, considers that detailed analysis of failures associated with HELB be analysed in detail to determine their potential impact on adjacent safety significant SSCs.

267 The verification and validation work has demonstrated that the expectations of the SAPs, specifically, SAP EHA.5, EHA.6, and EHA.14 have been addressed and that the expectations detailed within IAEA guidance have been met through the detailed and comprehensive analyses that have been undertaken by EDF and AREVA when considering the potential consequences of High Energy Line Break.

4.3.4 Internal Missile

268 The detailed assessment of the submission provided was included within the report, *"GDA Close-out for the EDF and AREVA UK EPR™ Reactor - GDA Issue **GI-UKEPR-IH-04 Revision 2** – Consequences of Missile Generation Arising from Failure of RCC-M Components"* (Ref. 25), in which it was explained that the expectations of the SAPs and international guidance have been met. However, it did identify the need for the barriers claimed as part of the internal missile safety case to be identified and as a result Reference 25 contains an assessment finding, **(AF-UKEPR-IH-08)**, to this effect.

5 REVIEW OF THE UPDATE TO THE PCSR

5.1 13.2. Internal Hazards

269 Sections 2, 4, 7, and 8 of Chapter 13.2 of the PCSR (Ref. 31) consider each of the areas of verification and validation required by the 4 GDA Issue Actions associated with this GDA Issue. The submission was reviewed to ensure that the outcome of the GDA assessment had been appropriately captured therein.

270 There have been changes to the PCSR arising from the assessment that has been undertaken associated with internal flooding, specifically in relation to structural integrity claims for pipework detailed within Section 2. The change is closely linked with the assessment done in support of the resolution of **GI-UKEPR-IH-03** relating to internal flooding and has resulted in claims associated with break preclusion being linked through to Chapters 5.2 and 10.5 of the PCSR.

271 I am satisfied that the updated PCSR reflects the findings from the GDA and the text has been updated to reflect the GDA assessment undertaken for close out of this GDA Issue.

6 ASSESSMENT FINDINGS

6.1 Additional Assessment Findings

272 The following assessment findings have been raised that requires to be resolved during the site specific phase:

> ***AF-UKEPR-IH-10**: The Licensee shall ensure that the design changes arising from the Flamanville 3 Verification and Validation process are appropriately considered within the site specific design.*
>
> ***Required Timescale:** "Mechanical, Electrical, and C&I Safety Systems – Before inactive commissioning".*
>
> ***AF-UKEPR-IH-11:** The Licensee shall ensure that the further analysis of the options to prevent the spread of flood water from the adjacent technical galleries resulting in loss of safety classified plant and equipment within the Diesel Generator Buildings are captured as part of the site specific design.*
>
> ***Required Timescale:** "Mechanical, Electrical, and C&I Safety Systems – Before inactive commissioning".*
>
> ***AF-UKEPR-IH-12:** The Licensee shall provide a means by which to physically identify individual safety classified cables from different safety divisions within UKEPR™ to ensure that there is visual identification of the different safety class cables for each division on plant.*
>
> ***Required Timescale:** "Mechanical, Electrical, and C&I Safety Systems – Before inactive commissioning".*
>
> ***AF-UKEPR-IH-13:** The Licensee shall ensure that the further analyses required arising from the Flamanville 3 Verification and Validation process are appropriately considered within the site specific design.*
>
> ***Required Timescale:** "Mechanical, Electrical, and C&I Safety Systems – Before inactive commissioning".*

6.2 Impacted Step 4 Assessment Findings

273 No assessment findings raised during Step 4 have been impacted as a result of this assessment.

7 ASSESSMENT CONCLUSIONS

274 Further to the GDA Issue, **GI-UKEPR-IH-02** and receipt of the deliverables detailed within the Resolution plan together with the responses to the TQs raised, I am satisfied that the approach taken to the verification and validation for the internal hazards described within this GDA Issue has been shown to be to a good standard.

275 My judgement of the adequacy of the response to the GDA issue is based upon the following factors:

- The approach to the analyses provided for each of the areas in which verification and validation was sought has been comprehensive and robust.
- The analyses themselves have demonstrated that the potential for loss of more than one redundancy is low and where the potential exists for more than one redundancy to be threatened there are robust operational arguments presented to demonstrate that loss of more than one redundancy would be tolerable and nuclear safety would not be compromised. In some cases diverse means by which nuclear safety can be assured have been presented as part of the analysis.
- The submissions provided in response to the GDA issue together with the PCSR (Ref. 11) have shown to align with my expectations in relation to standards, guidance, and relevant good practice.
- There have been 4 assessment findings raised as a result of my assessment, however, they are largely due to the need for the future licensee to capture the findings and approach taken for the verification and validation that has been undertaken for FA3 and address aspects of the findings presented.

276 I am, therefore, satisfied that GDA Issue, **GI-UKEPR-IH-02**, can now be closed.

8 REFERENCES

1 *ONR HOW2. Permissioning - Purpose and Scope of Permissioning.* PI/FWD, Issue 3. HSE. August 2011.

2 *Safety Assessment Principles for Nuclear Facilities.* 2006 Edition Revision 1. HSE. January 2008. www.hse.gov.uk/nuclear/SAP/SAP2006.pdf.

3 *Deterministic Safety Analysis and the Use of Engineering Principles in Safety Assessment.* T/AST/006 Issue 03, HSE, July 2000.

Internal Hazards. T/AST/014 Issue 02. HSE, August 2008.

Structural Integrity Civil Engineering Aspects. T/AST/017 Issue 02. HSE. March 2005

Diversity, Redundancy, Segregation and Layout of Mechanical Plant. T/AST/036 Issue 02, HSE, June 2009.

Guidance on the Purpose, Scope and Content of Nuclear Safety Cases. T/AST/051 Issue 01, HSE, May 2002

www.hse.gov.uk/nuclear/operational/tech_asst_guides/index.htm.

4 *Western European Nuclear Regulators' Association. Reactor Harmonization Group. WENRA Reactor Reference Safety Levels.* WENRA. January 2008. www.wenra.org.

5 *Safety of Nuclear Power Plants: Design. Safety Requirements.* International Atomic Energy Agency (IAEA). Safety Standards Series No. NS-R-1. IAEA. Vienna. 2000.

Protection against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants. International Atomic Energy Agency (IAEA), Safety Guide, NS-G-1.11, IAEA, Vienna 2004.

Protection against Fires and Explosions in the Design of Nuclear Power Plants. International Atomic Energy Agency (IAEA), Safety Guide, NS-G-1.7, IAEA, Vienna 2004.

www.iaea.org.

6 *GDA Issue GI-UKEPR-IH-02 Revision 2.* ONR. July 2011. TRIM Ref. 2011/361937. (in TRIM folder 5.1.3.6348.)

7 *Step 4 Internal Hazards Assessment of the EDF and AREVA UK EPR™ Reactor*. ONR Assessment Report ONR-GDA-AR-11-001 Revision 0. TRIM Ref. 2010/581514. (in TRIM folder 4.4.1.1827.).

8 *Resolution Plan for GDA Issue GI-UKEPR-IH-02 Revision 2.* EDF and AREVA. June 2011. TRIM Ref. 2011/256665. (in TRIM folder 5.1.3.6351.)

9 *Reference Design Configuration*. UKEPR-I-002 Revision 13. UK EPR. September 2012. TRIM Ref. 2012/350053.

10 *Design Change Procedure.* UKEPR-I-003 Revision 9. EDF and AREVA. June 2012. TRIM Ref. 2012/243501.

11 *UK EPR GDA Step 4 Consolidated Pre-construction Safety Report – March 2011.* EDF and AREVA. Detailed in EDF and AREVA letter UN REG EPR00997N. 18 November 2011. TRIM Ref. 2011/552663.

12 *Internal Hazards Assessment Plan for GDA Close out of the EDF and AREVA UKEPR™.* ONR, 26 September 2011. TRIM Ref. 2011/560168

13 *EDF and AREVA UK EPR™ - Schedule of Technical Queries Raised during GDA Step 1 to Step 4.* HSE-ND. TRIM Ref. 2010/600726.

14 *EDF and AREVA UK EPR™ - Schedule of Technical Queries Raised during GDA Close-out.* Office for Nuclear Regulation. TRIM Ref. 2011/389411.

15 *Flooding Safety Analysis for Nuclear Auxiliary Building HNX.* PF2009EN0001, Revision D, February 2011, EDF. TRIM Ref. 2011/324015.

16 *Flooding event: Analysis of the Safeguard Building.* EZT2009EN0005, Revision E, March 2011, EDF. TRIM Ref. 2011/324002

17 *EPR FA3 Fuel Building Internal Flooding Assessment*. EYRT2009FR0008 Revision C1, November 2011, EDF. TRIM Ref. 2011/635079.

18 *Study of Internal Flooding Events in the Diesel Generator Building of the Flamanville 3 EPR.* EYRT/2009/FR/0032 Revision E, April 2012, EDF. TRIM Ref. 2012/180235

19 *Analysis of Internal Flooding within the EPR FA3 Reactor Building.* EYRT/2009/FR/0076 Revision D, July 2011, EDF. TRIM Ref. 2011/635074

20 *Location of cable trays in the FA3 EPR Nuclear Island.* EYRT2010FR0042 Revision B, November 2011, EDF. TRIM Ref. 2011/635073.

21 *Application of the MTE175 "Breaches of High Energy Pipes, load caused on the structural steelworks".* ECEIG091393 Revision A1, May 2011, EDF. TRIM Ref. 2011/324000.

22 *Additional information for the stage 1 analysis report on the consequences of high energy line breaks – fuel building.* ECEF101595 Revision A1, May 2011, EDF. TRIM Ref. 2011/323998.

23 *Additional information for the stage 1 analysis report on the consequences of high energy line breaks – safeguard auxiliary buildings.* ECEF101596 Revision A1, May 2011, EDF. TRIM Ref. 2011/324013.

24 *EPR Internal Missiles – Risk assessment report on building structure and layout.* ECEIG091634, Revision B1, EDF, April 2011. TRIM Ref. 2011/324014

25 *GDA Close-out for the EDF and AREVA UK EPR™ Reactor - GDA Issue GI-UKEPR-IH-04 Revision 2 – Consequences of Missile Generation Arising from Failure of RCC-M Components.* ONR Assessment Report ONR-GDA-AR-12-015 Revision 0. TRIM Ref. 2012/15.

26 *GDA Issue GI-UKEPR-IH-03 Revision 2.* ONR. July 2011. TRIM Ref. 2011/360465. (in TRIM folder 5.1.3.6348.)

27 *GDA Close-out for the EDF and AREVA UK EPR™ Reactor - GDA Issue GI-UKEPR-IH-03 Revision 2 – Internal Flooding Safety Case.* ONR Assessment Report ONR-GDA-AR-12-018 Revision 0. TRIM Ref. 2012/18.

28 *Treatment of open points from the 1st phase HELB study in the Fuel Building.* EYRT2010FR0050, Revision B1, December 2011, EDF. TRIM Ref. 2011/652615.

29 *High Energy Line Break: Treatment of open points – common mode failure and other safety related aspects in the Safeguards Buildings.* EZLT2010EN0005, Revision B, August 2010, EDF. TRIM Ref. 2011/556408.

30 *1st Stage Analysis: consequences of high energy line breaks – safeguard auxiliary and electrical buildings.* ECEF092042, Revision A, February 2010, EDF. TRIM Ref. 2011/85907.

31 PCSR Sub-Chapter 13.2 Update – Internal Hazards Protection, UKEPR-0002-132 Issue 05, 31st October 2012, TRIM 2012/450702.

Table 1

Relevant Safety Assessment Principles Considered for Close-out of GI-UKEPR-IH-02 Revision 2

SAP No.	SAP Title	Description
SC.4	Safety case characteristics	A safety case should be accurate, objective and demonstrably complete for its intended purpose.
EKP.3	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.
EKP.4	Safety function	The safety function(s) to be delivered within the facility should be identified by a structured analysis.
EKP.5	Safety Measure	Safety measures should be identified to deliver the required safety function(s).
ECS.1	Safety Categorisation	The safety functions to be delivered within the facility, both during normal operation and in the event of a fault or accident, should be categorised based on their significance with regard to safety.
ECS.2	Safety classification of structures, systems and components	Structures, systems and components that have to deliver safety functions should be identified and classified on the basis of those functions and their significance with regard to safety.
EDR.2	Redundancy, diversity and segregation	Redundancy, diversity and segregation should be incorporated as appropriate within the designs of structures, systems and components important to safety.
EDR.4	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.

Table 1

Relevant Safety Assessment Principles Considered for Close-out of GI-UKEPR-IH-02 Revision 2

SAP No.	SAP Title	Description
ELO.4	Minimisation of the effects of incidents	The design and layout of the site and its facilities, the plant within a facility and support facilities and services should be such that the effects of incidents are minimised.
EHA.1	Identification	External and internal hazards that could affect the safety of the facility should be identified and treated as events that can give rise to possible initiating faults.
EHA.3	Design basis events	For each internal or external hazard, which cannot be excluded on the basis of either low frequency or insignificant consequence, a design basis event should be derived.
EHA.4	Frequency of exceedance	The design basis event for an internal and external hazard should conservatively have a predicted frequency of exceedance in accordance with the fault analysis requirements (FA.5).
EHA.5	Operating conditions	Hazard design basis faults should be assumed to occur simultaneously with the most adverse normal facility operating condition.
EHA.6	Analysis	Analyses should take into account simultaneous effects, common cause failure, defence in depth and consequential effects.
EHA.7	‘Cliff-edge’ effects	A small change in DBA parameters should not lead to a disproportionate increase in radiological consequences.
EHA.10	Electromagnetic interference	The design of facility should include protective measures against the effects of electromagnetic interference.
EHA.13	Fire, explosion, missiles, toxic gases etc – use and storage of hazardous materials	The on-site use, storage or generation of hazardous materials should be minimised, and controlled and located so that any accident to, or release of, the materials will not jeopardise the establishing of safe conditions on the facility.

Table 1

Relevant Safety Assessment Principles Considered for Close-out of GI-UKEPR-IH-02 Revision 2

SAP No.	SAP Title	Description
EHA.14	Fire, explosion, missiles, toxic gases etc – sources of harm	Sources that could give rise to fire, explosion, missiles, toxic gas release, collapsing or falling loads, pipe failure effects, or internal and external flooding should be identified, specified quantitatively and their potential as a source of harm to the nuclear facility assessed.
EHA.15	Fire, explosion, missiles, toxic gases etc – effects of water	The design of the facility should prevent water from adversely affecting structures, systems and components important to safety.
EHA.16	Fire, explosion, missiles, toxic gases etc – fire detection and fighting	Fire detection and fire-fighting systems of a capacity and capability commensurate with the credible worst-case scenarios should be provided.
FA.6	Fault sequences	For each initiating fault in the design basis, the relevant design basis fault sequences should be identified.

Annex 1

Deliverables and Associated Technical Queries Raised During Close-out Phase

GI-UKEPR-IH-02 Revision 2 – Verification and Validation – EDF and AREVA Deliverables

GDA Issue Action	Internal Hazards Topic	Document Ref.	Title	Ref.
GI-UKEPR-IH-02.A1	Internal Flooding	PF2009EN0001 Revision D	HN flooding analysis: “Flooding Safety Analysis for Nuclear Auxiliary Building”	Ref. 15
GI-UKEPR-IH-02.A1	Internal Flooding	EZT2009EN0005 Revision E	SAB flooding analysis: “Flooding event – Analysis of the Safeguard Building”	Ref. 16
GI-UKEPR-IH-02.A1	Internal Flooding	EYRT2009FR0008 Revision C	EPR FA3 Fuel Building Internal Flooding Assessment.	Ref. 17
GI-UKEPR-IH-02.A1	Internal Flooding	EYRT2009FR0032 Revision E.	Study of Internal Flooding Events in the Diesel Generator Building of the Flamanville 3 EPR.	Ref. 18
GI-UKEPR-IH-02.A1	Internal Flooding	EYRT2009FR0076 Revision D	Analysis of internal flooding in the FA3 EPR reactor building.	Ref. 19
GI-UKEPR-IH-02.A2	Internal Fire	EYRT2010FR0042 Revision B	Location of cable trays in the FA3 EPR Nuclear Island.	Ref. 20
GI-UKEPR-IH-02.A3	High Energy Line Break	ECEIG091393 Revision A1	Application of the MTE175 "Breaches of High Energy Pipes, load caused on the structural steelworks".	Ref. 21
GI-UKEPR-IH-02.A3	High Energy Line Break	ECEF101595 Revision A1	Additional information for the stage 1 analysis report on the consequences of high energy line breaks – fuel building.	Ref. 22

Annex 1

Deliverables and Associated Technical Queries Raised During Close-out Phase

GI-UKEPR-IH-02 Revision 2 – Verification and Validation – EDF and AREVA Deliverables

GDA Issue Action	Internal Hazards Topic	Document Ref.	Title	Ref.
GI-UKEPR-IH-02.A3	High Energy Line Break	ECEF101596 Revision A1	Additional information for the stage 1 analysis report on the consequences of high energy line breaks – safeguard auxiliary buildings.	Ref. 23
GI-UKEPR-IH-02.A4	Internal Missile	ECEIG091634, Revision B1	EPR Internal Missiles – Risk assessment report on building structure and layout.	Ref. 24

GI-UKEPR-IH-02 Revision 2 – Verification and Validation – Technical Queries Raised

TQ Reference	GDA Issue Action	Related Submission	Description
TQ-EPR-1472	**GI-UKEPR-IH-02.A1**	EZT2009EN0005 Revision E	*Reference request from the Safeguards Auxiliary Building flooding analysis: EZT2009EN0005 “Flooding event – Analysis of the Safeguard Building”*
TQ-EPR-1474	**GI-UKEPR-IH-02.A1**	EZT2009EN0005 Revision E	*Further Claims made on doors and barriers to protect against internal flooding.*
TQ-EPR-1475	**GI-UKEPR-IH-02.A1**	EZT2009EN0005 Revision E	*Safety Case impact of recommendations contained within verification and validation reports.*
TQ-EPR-1503	**GI-UKEPR-IH-02.A3**	ECEF101596 Revision A1	*Reference request - “EZLT2010EN0005B, High Energy Line Breaks: Treatment of the outstanding points – common mode failure and other safety related aspects in the Safeguard Buildings"*

Annex 1

Deliverables and Associated Technical Queries Raised During Close-out Phase

GI-UKEPR-IH-02 Revision 2 – Verification and Validation – Technical Queries Raised

TQ Reference	GDA Issue Action	Related Submission	Description
TQ-EPR-1514	**GI-UKEPR-IH-02.A3**	ECEF101595 Revision A1	*Reference request - “EYRT2010EN0050B, High Energy Line Breaks: Treatment of the outstanding points – common mode failure and other safety related aspects in the Fuel Building" and "EYRT2010FR0041A, Identification of cable raceways routing through HELB rooms in a foreign division in the BR and the BK".*

Annex 2

GDA Assessment Findings Arising from GDA Close-out for Internal Hazards GDA Issue GI-UKEPR-IH-02

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-IH-10	The Licensee shall ensure that the design changes arising from the Flamanville 3 Verification and Validation process are appropriately considered within the site specific design.	*“Mechanical, Electrical, and C&I Safety Systems – Before inactive commissioning”*.
AF-UKEPR-IH-11	The Licensee ensure that the further analysis of the options to prevent the spread of flood water from the adjacent technical galleries resulting in loss of safety classified plant and equipment within the Diesel Generator Buildings are captured as part of the site specific design.	*“Mechanical, Electrical, and C&I Safety Systems – Before inactive commissioning”*.
AF-UKEPR-IH-12	The Licensee shall provide a means by which to physically identify individual safety classified cables from different safety divisions within UKEPR™ to ensure that there is visual identification of the different safety class cables for each division on plant.	*“Mechanical, Electrical, and C&I Safety Systems – Before inactive commissioning”*.
AF-UKEPR-IH-13	The Licensee shall ensure that the further analyses required arising from the Flamanville 3 Verification and Validation process are appropriately considered within the site specific design.	*“Mechanical, Electrical, and C&I Safety Systems – Before inactive commissioning”*.

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 3

GDA Issue, GI-UKEPR-IH-02 – Internal Hazards – UK EPR™

EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT

GDA ISSUE

VERIFICATION AND VALIDATION

GI-UKEPR-IH-02 REVISION 2

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Fault Studies
PSA</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-UKEPR-IH-02</td><td>GDA Issue Action Reference</td><td>GI-UKEPR-IH-02.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">Outstanding Verification and Validation for internal flooding, cable routing, high energy line break and missiles forms part of the requisite evidence and will be required in order to demonstrate an adequate internal hazards safety case.</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide the requisite evidence in the form of the detailed Flamanville 3 verification and validation analysis and/or other supporting documentation in support of the claims and arguments presented within Chapter 13.2 of the PCSR associatedwith internal flooding. The response should include analysis that supports the claims and arguments relating to:
Civil structures (including surface coatings) claimed as flood barriers.
Watertight doors and penetrations including qualification data.
Drains and sumps claimed to prevent damage to nuclear significant SSCs.
Calculations in place to support any claims made on potential water volumes.
Any further defence in depth and ALARP measures that could be implemented into the design.
Any identified design changes and their implementation within the PCSR.
The list above should not be considered to be exhaustive and the items detailed above are provided as a means to inform EDF and AREVA of my expectations.
With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

Annex 3

GDA Issue, GI-UKEPR-IH-02 – Internal Hazards – UK EPR™

EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT

GDA ISSUE

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GI-UKEPR-IH-02 REVISION 2

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<tr><td>GDA Issue Action</td><td colspan="3">Provide the requisite evidence in the form of the detailed Flamanville 3 verification and validation analysis and/or other supporting documentation in support of the claims and arguments presented within Chapter 13.2 of the PCSR associated with the routing of electrical cables within the EPR design in order to prevent a single fire resulting in loss of more than one divisional separation group.
The response should include analysis that supports the claims and arguments relating to:
• The routing and identification of protected cable trays.
• Justification of claims and arguments made relating to geographical separation.
• The provision of passive protection applied to cables and cable trays specifically.
• Any further defence in depth and ALARP measures that could be implemented into the design.
• Any identified design changes and their implementation within the PCSR.
The list above should not be considered to be exhaustive and the items detailed above are provided as a means to inform EDF and AREVA of my expectations.
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GDA Issue, GI-UKEPR-IH-02 – Internal Hazards – UK EPR™

EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT

GDA ISSUE

VERIFICATION AND VALIDATION

GI-UKEPR-IH-02 REVISION 2

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<tr><td>GDA Issue Reference</td><td>GI-UKEPR-IH-02</td><td>GDA Issue Action Reference</td><td>GI-UKEPR-IH-02.A3</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide the requisite evidence in the form of the detailed Flamanville 3 verification and validation analysis, specifically, the FA3 1st Stage Pipe Break Analysis and/or other supporting documentation in support of the claimsand arguments presented within Chapter 13.2 of the PCSR associated with high energy line break (HELB) within the EPR design. The response should include analysis that supports the claims and arguments relating to:
• Consequence analysis, where applicable.
• Break preclusion.
• Identification and qualification of physical restraints, barriers and doors.
• Identification and qualification of pressure relief panels/routes.
• Any further defence in depth and ALARP measures that could be implemented into the design.
• Any identified design changes and their implementation within the PCSR.
The list above should not be considered to be exhaustive and the items detailed above are provided as a means to inform EDF and AREVA of my expectations.
With agreement from the Regulator this action may be completed by alternative means.</td></tr>
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Annex 3

GDA Issue, GI-UKEPR-IH-02 – Internal Hazards – UK EPR™

EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT
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<tr><td>GDA Issue Action</td><td colspan="3">Provide the requisite evidence in the form of the detailed Flamanville 3 verification and validation analysis and/or other supporting documentation in support ofthe claims and arguments presented within Chapter 13.2 of the PCSR associated with internal missiles. The response should include analysis that supports the claims and arguments relating to:
• Identification of all potential sources of internal missile which could result in a threat to nuclear safety significant SSCs.
• Consequence analysis, where applicable.
• Break preclusion.
• Identification and qualification of physical restraints, barriers and doors.
• Any further defence in depth and ALARP measures that could be implemented into the design.
• Any identified design changes and their implementation within the PCSR.
The list above should not be considered to be exhaustive and the items detailed above are provided as a means to inform EDF and AREVA of my expectations.
With agreement from the Regulator this action may be completed by alternative means.</td></tr>
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