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-01 Revision 2 – Substantiation and Analysis of the Consequences of Dropped Loads and Impact from Lifting Equipment included within the EPR Design.

Assessment Report: ONR-GDA-AR-12-016
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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-01 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 Steps 3 and 4 it became apparent that the arguments and evidence in support of the claims made associated with dropped loads and impact had not been presented. This was due to claims that discounted the potential for dropped loads and impact from Safety Class 1 lifting equipment and claims made on structures for Safety Class 2 lifting equipment. As a result the GDA Issue was raised which required EDF and AREVA to provide substantiation of the claims made within the PCSR associated with dropped loads and impact for the UK EPR™.

The approach taken by EDF and AREVA was to produce detailed consequence analyses for a number of potential lifting operations and aimed to demonstrate the risk to nuclear safety from a dropped load or impact was ALARP. These analyses 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 comprising of quantitative consequence analyses undertaken for dropped loads and impact arising from Safety Class 1 and Safety Class 2 lifting equipment, I am satisfied that the safety case for dropped loads and impact for the UK EPR™ is adequate.

My judgement is based upon the following factors:

- The approach to analyse the quantitative consequences of dropped loads and impact for Safety Class 1 and Safety Class 2 lifting equipment is in line with the HSE SAPs as well as internal guidance and relevant good practice.
- The analyses provided are comprehensive and have found that the consequences of a dropped load or impact from lifting equipment proposed for the UK EPR™ are acceptable to nuclear safety.
- The design of the lifting equipment is to a high standard and consistent with expectations within the United Kingdom and worldwide.
- The approach to the analysis of the consequences of failure together with the operating conditions is in line with the expectations of mechanical engineering assessors within ONR.
- The claims made associated with the civil structures have been subject to assessment by civil engineering assessors and found to be acceptable.
- EDF and AREVA have identified design changes as a result of the consequence analyses undertaken which, once implemented, will demonstrate that the provisions in place to protect against a dropped load or impact associated with Safety Class 1 and Safety Class 2 lifting equipment are ALARP.

One Assessment Finding has been raised in relation to this assessment, which requires a future Licensee to provide evidence associated with the further studies in order to support the design modification for the manual connection of the Low Head Safety Injection/Residual Heat Removal (LHSI/RHR) system following a Loss of Coolant Accident (LOCA) caused by a dropped load from the Polar Crane and demonstrate that the provisions in place are ALARP.

The Stage 2 Change Modification Forms (CMFs) associated with dropped loads and impact have been submitted to ONR. The two CMFs, CMF34 and CMF35, have been reviewed and I am satisfied that the outcome of the consequence analyses undertaken in response to this GDA Issue have been adequately captured.

The updated PCSR has been reviewed and I am satisfied that the outcome of the analyses undertaken has been adequately reflected therein.

I am, therefore, satisfied that GDA Issue, **GI-UKEPR-IH-01**, can now be closed.

LIST OF ABBREVIATIONS

ALARP	As Low As Reasonably Practicable
AREVA	AREVA NP SAS
CMF	Change Modification Form
CRDM	Control Rod Drive Mechanism
CVCS	Chemical and Volume Control System
DAC	Design Acceptance Confirmation
EDF	Electricité de France SA
FB	Fuel Building
GDA	Generic Design Assessment
HSE	Health and Safety Executive
IAEA	International Atomic Energy Agency
LHSI	Low Head Safety Injection
LOCA	Loss of Coolant Accident
LOLER	Lifting Operations and Lifting Equipment Regulations
MSTM	Multi-Stud Tensioning Machine
ONR	Office for Nuclear Regulation (an agency of HSE)
PCC	Plant Condition Category
PCSR	Pre-construction Safety Report
PLC	Programmable Logic Controller
RB	Reactor Building
RCCA	Rod Cluster Control Assembly
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RM	Refuelling Machine
RPV	Reactor Pressure Vessel
SAPs	HSE Safety Assessment Principle(s)
SFMB	Spent Fuel Mast Bridge
SQEP	Suitably Qualified and Experienced Person
SSC	Systems, Structures and Components
TAG	Technical Assessment Guide(s)
TQ	Technical Query
TSC	Technical Support Contractor

LIST OF ABBREVIATIONS

UK EPR™	EDF and AREVA UK specific pressurised water reactor design

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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-01 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 Plan (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. GDA Issues are unresolved issues considered by regulators to be significant, but resolvable, and which require resolution before nuclear island safety related construction of such a reactor could be considered. Assessment Findings are findings that are identified during the regulators' GDA assessment that are important to safety, but not considered critical to the decision to start nuclear island safety related construction of such a reactor.

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-01**.

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™ (Ref. 7) 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 Issues highlight shortfalls not previously identified during Step 4, there will be a need for these aspects of the assessment to be 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 Issues may leave aspects of the assessment requiring further detailed evidence when the information becomes available at a later stage.

8 During Steps 3 and 4 it became apparent that the arguments and evidence in support of the claims made associated with dropped loads and impact were not in line with ONR expectations. This was due to claims associated with discounting the potential for

dropped loads and impact from Safety Class 1 lifting equipment as well as claims made on structures for Safety Class 2 lifting equipment.

9 The Step 3 Internal Hazards Assessment Report for the EDF and AREVA UK EPR (Ref. 28) stated:

"Further evidence of the adequacy of the approach to the methodology applied to the identification of dropped loads and internal missiles should be further investigated during Step 4 when the two outstanding documents are supplied."

10 This task was subsequently undertaken during Step 4 (Ref. 7) and the assessment found that the approach to the claims made on the highest integrity lifting equipment (Safety Class 1 lifting equipment) were not in line with our expectations as they discounted the potential for a dropped load or impact as a result of failure or operator actions. As a result of claiming that load drops were precluded no analysis of the consequences of failure were undertaken.

11 There are also lifts of nuclear safety significance undertaken by lifting equipment that is not designed to preclude dropped loads (Safety Class 2 lifting equipment). The principal claim for dropped loads and impacts associated with this lifting equipment were civil engineering aspects of the construction and the tolerability of a dropped load on to an area. Once again, there was no analysis of the potential consequence of the dropped load or impact provided and the SAP (Ref. 2), SAP EHA.14 would expect a consequence analysis be undertaken, be that qualitative or quantitative. As a result the GDA Issue included the need to undertake a consequence analysis for a number of lifts associated with the Safety Class 2 lifting equipment. EDF and AREVA have selected a number of representative lifts and assessed the potential consequences in the event of a dropped load or impact. In addition, the GDA Issue includes an action associated with the civil engineering claims on the structures in the event of a dropped load or impact given that this formed a significant leg of the safety case.

12 As a result the GDA Issue (Ref. 6) was raised which required EDF and AREVA to provide substantiation of the claims made within the March 2011 Consolidated PCSR (Ref. 12) associated with dropped loads and impact for the UK EPR™.

1.3 Methodology

13 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.

14 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.

15 The assessment allows ONR to judge whether the submissions provided in response to the GDA Issue are sufficient to allow it to be closed. Where requirements for more detailed evidence have been identified that are appropriate to be provided at the design, construction or commissioning phases of the project these can be carried forward as Assessment Findings.

1.4 Structure

16 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 of the EDF and AREVA UK EPR™ (Ref. 7). 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.

17 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

18 The intended assessment strategy for GDA Close-out for the internal hazards topic area was set out in an Assessment Plan (Ref. 13) that identified the intended scope of the assessment and the standards and criteria that would be applied.

19 The overall basis for the assessment of the GDA Issues are the internal hazards elements of:

- Submissions made to ONR in accordance with the Resolution Plan (Ref. 8).
- 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 and includes Change Management Forms (CMF).

2.1 The Approach to Assessment for GDA Close-out

20 The approach to the closure of GDA Issue for the UK EPR™ Project involves:

- Assessment of submissions made by EDF and AREVA in response to the GDA Issue identified through the GDA process. These submissions are detailed within the EDF and AREVA Resolution Plan for the GDA Issue.

21 If the assessment of the submissions together with any design changes requested by EDF and AREVA are judged acceptable, the GDA Issue can be cleared.

2.2 Standards and Criteria

22 The relevant standards and criteria adopted within this assessment are principally the Safety Assessment Principles (SAP) (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

23 The key SAPs 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

24 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/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.

- T/AST/056 Issue 02 – Nuclear Lifting Operations.

2.2.3 International Standards and Guidance

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

- 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-G-1.11 (Ref. 5)

2.3 Use of Technical Support Contractors

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

2.4 Out-of-scope Items

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

3 EDF AND AREVA DELIVERABLES IN RESPONSE TO THE GDA ISSUE

28 In response to the GDA Issue, EDF and AREVA provided a Resolution Plan (Ref. 8) detailing how they intended to address the GDA Issue Actions.

29 The Resolution Plan that dealt with **GI-UKEPR-IH-01.A1** stated:

"EDF/AREVA will provide substantiation of the nuclear safety significant structures, systems and components vulnerable to dropped load and impact from RS1 and RS2 lifting equipments."

30 EDF and AREVA stated that the substantiation would involve the production of a safety case covering representative dropped loads and demonstrate that the provisions in place to ensure that the risk to nuclear safety of a dropped load or impact was ALARP. As part of the case they confirmed that it would consider:

- Claims on civil structures and additional physical protection.
- Limits and conditions on the use of the RS1 and RS2 lifting equipment.
- Provision of detailed load path routes.
- Measures in place to ensure that the potential for impact of the load is minimised.
- Any identified design changes
- Any further defence in depth and ALARP measures that could be implemented into the design.

31 The Resolution Plan that dealt with **GI-UKEPR-IH-01.A2** stated:

"A methodology report will be produced defining the approach for evaluating the dropped loads consequences on Civil Structures and Steel Structures. The methodology will be consistent with requirements of ETC-C AFCEN."

32 EDF and AREVA committed to provide a methodology to treat dropped loads and impacts on civil structures which would include:

- Derivation of design loads.
- Analysis methods.
- Design rules.
- Reliability expectations.
- Global stability considerations.

33 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	Dropped Loads and Impact	Deliverable	Ref.
GI-UKEPR-IH-01.A1	Summary of Design Basis and Principles	*Dropped Loads – Summary of Design Basis and Principles,* ECEIG111683, Revision A.	16
GI-UKEPR-IH-01.A1	Justification of the Dropped Load Cases selected for detailed Study	*Identification of Representative Drop Load Cases from the Safety Class 1 Polar Crane in the Reactor Building,*	17

GDA Issue Action	Dropped Loads and Impact	Deliverable	Ref.
		PEPS-G/2011/en/1060, Revision A.	
GI-UKEPR-IH-01.A1	Justification of the Dropped Load Cases selected for detailed Study	*Identification of Representative Drop Load Cases from the Safety Class 2 Cranes*, ECEIG111791, Revision A.	18
GI-UKEPR-IH-01.A1	Summary report and supporting documents	*Summary Report for the Substantiation of "Dropped Loads" Hazard*, ECEIG120274, Revision A.	19
GI-UKEPR-IH-01.A1	Summary report and supporting documents	*Safety Case for 4 Representative Load Drops from Safety Classified 2 Cranes,* ECEIG120198, Revision A.	20
GI-UKEPR-IH-01.A1	Summary report and supporting documents	*Application Note for a Drop Load Impact on a Reinforced Concrete Slab,* ECEIG111395, Revision A.	21
GI-UKEPR-IH-01.A1	Summary report and supporting documents	*EPR UK – RS2 cranes – Drop Load Impact Calculations,* ECEIG111620, Revision A.	22
GI-UKEPR-IH-01.A1	Summary report and supporting documents	*ALARP justification of Identified Representative Drop Load Cases from the Safety Class 1 Polar Crane in the Reactor Building,* PEPS-G/2011/en/1076, Revision C.	23
GI-UKEPR-IH-01.A1	Summary report and supporting documents	*Consequences on the Reactor of an Accidental RPV Head Drop During it's Handling,* PEER-F DC 71, Revision B.	24
GI-UKEPR-IH-01.A1	Summary report and supporting documents	*Drop of a Reactor Cavity Cover Slab on the RPV Closure Head Analysis,* PEPR–F DC 85, Revision B.	25
GI-UKEPR-IH-01.A1	Summary report and supporting documents	*Check of Bearing Capacity of Reinforced Concrete Reactor Pool Slab Subject to Drop Load of a Concrete Cover Slab and a Multi-Stud Tensioning Machine,* PECS-G/2011/en/0018, Revision B.	26
GI-UKEPR-IH-01.A2	Methods with regard to the risk of dropped loads for UK EPR for concrete structure	*Methods with regard to the risk of dropped loads for UK EPR for concrete structure,* ENGSGC100483, Revision B	27

34 An overview of References 16-19 is provided within this section as Reference 19 provides a summary of the extensive work that has been undertaken to underpin the safety case for dropped loads and impact detailed within References 20-26.

35 The submission provided in response to GDA Issue Action, **GI-UKEPR-IH-01.A2** is primarily associated with the civil engineering claims associated with dropped loads and impact. The assessment of this submission was undertaken by Civil Engineering Assessors. The overview of the submission and subsequent assessment is therefore not detailed within this internal hazards assessment; however, the conclusions arising from the assessment are detailed together with a reference to the civil engineering assessment.

36 It is important to note that this information is supplementary to the information provided within the March 2011 Consolidated PCSR (Ref. 12) which has already been subject to assessment during earlier stages of GDA. In addition, the deliverables are not intended to provide the complete safety case for the dropped load and impact hazard. Rather they form further detailed arguments and evidence to supplement those already provided during earlier Steps within the GDA Process.

37 The deliverables associated with this GDA Issue use the existing French approach to classification and categorisation of Structures, Systems, and Components (SSCs). The use of categorisation and classification is addressed as part of the work undertaken in response to the cross cutting GDA Issue, **GI-UKEPR-CC-01**.

38 The definition of RS1 and RS2 lifting equipment cited within the GDA Issue and the associated Resolution Plan are broadly equivalent to Safety Class 1 and Safety Class 2 lifting equipment. As a result where a submission has referred to RS1 or RS2 this can be read as being the same as Safety Class 1 and Safety Class 2 from a UK regulatory perspective. In addition, RS1 and RS2 are equivalent to *higher requirements* and *additional requirements*, respectively, as defined within Section 3.2 of the March 2011 Consolidated PCSR (Ref. 12)

3.1 Dropped Loads – Summary of Design Basis and Principles, ECEIG111683 Revision A

39 The above submission (Ref. 16) details the dropped load hazard studies that have been undertaken by EDF and AREVA. It provides a description of the classification principles for lifting and handling devices together with information on the Safety Class 1 and Safety Class 2 lifting equipment analysed including load path routes. The report summarises the acceptance criteria as well as the prevention measures in place based on existing EDF nuclear plants. In addition information is presented associated with operating experience feedback.

40 The report identifies a number of UK standards relevant to lifting operations and equipment associated with operational facilities, e.g. The Lifting Operations and Lifting Equipment Regulations 1998 (LOLER) (Ref. 29) and The Management of Health and Safety at Work Regulations 1992 (Ref. 30). The report identifies a number of standards associated with the design of lifting equipment with a specific focus on high integrity lifting equipment used in existing nuclear facilities within France and Germany.

41 The standards applied to the design of the lifting equipment, namely the Nuclear Safety Standards Commission (KTA) standard, KTA 3902, "Design of Lifting Equipment in Nuclear Power Plants" (Ref. 31) and the EDF standard, BTS 60.C.007.03, "High Safety Lifting and Handling Machines" (Ref. 32), identify the lifting equipment as either requiring *"additional requirements"* or *"higher requirements"*. *"Additional requirements"* assume that

the load could drop whereas *“higher requirements”* do not assume load drop and the arguments are based upon a combination of engineered and administrative measures.

42 The lifting devices are classified into 3 categories:

- Safety Class 1 – this is applied to lifting equipment where consequences of failure are considered to be unacceptable and correspond to the classification of having *“higher requirements”*.
- Safety Class 2 – this is applied to lifting equipment where the consequences are considered to be serious and correspond to the classification of having *“additional requirements”*.
- Safety Class 3 – this is applied to lifting equipment where the consequences are less than those in Safety Class 2 and are not considered as part of the design basis.

43 The submission provides details of the classification of all the Safety Class 1 and Safety Class 2 lifting equipment including an overview of the engineered protection systems in place e.g. interlocks and limit switches to prevent unauthorised/unacceptable load movements. It also includes reference to the use of detailed load paths to ensure that the routes by which operators transport items using lifting equipment is ALARP. It emphasises that load drops from Safety Class 1 lifting equipment are discounted due to the high safety class of the lifting equipment. However, as the UK approach requires that a deterministic assessment is performed of the potential radiological consequences of all dropped loads that could be of nuclear safety significance, the submission states that, in the frame of GDA, load drops from all cranes should be considered.

44 The submission states that despite the high design standards applied to Safety Class 1 cranes, dedicated safety cases covering representative dropped loads from Safety Class 1 and Safety Class 2 lifting equipment will be produced in order to demonstrate that the provisions in place to ensure that the risk to nuclear safety of a dropped load or impact are ALARP. These cases are identified within two further submissions associated with GDA Issue Action, **GI-UKEPR-IH-01.A1** (see 3.2 and 3.3 below).

45 If the subsequent studies undertaken for the Safety Class 1 and Safety Class 2 lifting equipment identify that unacceptable consequences could occur, then EDF and AREVA will develop:

- Safety analysis in the form of an ALARP case (prevention measures, justification, modifications of the design).
- A radiological consequence study, noting that in order to simplify the study, bounding cases could be used.

46 In addition to the additional safety case analyses identified above, the submission provides the outcome of a review undertaken relating to operating feedback. The review sourced information within the French nuclear fleet and internationally as well as used WANO and NUREG reports. The principal conclusions from the review were:

- Most events did not but might have affected safety and had more significant consequences, but all could have been avoided. The shortfalls were associated with the human performance aspects of the work being undertaken with failures due to handling procedures, knowledge and skills of staff, deficiencies in the inspection, maintenance and verification of lifting. In addition, inappropriate practices, a lack of knowledge of the potential and real hazards, and control and supervision of site employees that does not allow effectively the identification and correction of safety problems. None of the events were associated with failures attributable to the design specifically.

- There was operating experience feedback from the US that showed that over half of drops occurred in the fuel pool without radiological consequences. They also identified human errors tend to increase over time due to the lack of, or non-compliance with, lifting procedures. Each of the three events in which there was a drop of a heavy load was attributable to failures of the slinging equipment and not with failures in the design of the lifting equipment involved.

47 EDF and AREVA conclude that the most of the events associated with dropped loads and impact are related to inappropriate lifting procedures, problems of operator training, maintenance or control, etc. Further, it is confirmed that the design of lifting and handling equipment of EDF Plants is not affected by taking into account the feedback on dropped loads in operation. This position is supported:

- by the absence of events impacting safety, related to crane failures, and,
- by reliability levels high enough for EPR, achieved through more stringent provisions for cranes and lifting beams, taking account of human factors (including the quality of operating procedures, staff training and the feedback experience).

48 Further information is provided relating to the existing prevention measures adopted at EDF Plants associated with both the design and administrative controls. The report identifies that measures have been put in place in all cases where the design has been specified; however, in a number of areas rigging arrangements have not yet been determined. The report refers to a detailed analysis that was undertaken arising from Regulatory Observation, (**RO-UKEPR-52)** (Ref. 33), raised during Step 4 of the GDA by the Mechanical Engineering Assessment Assessors. The analysis provided, *"UK EPR GDA – Management of Nuclear Safety Significant Lifting"*, (Ref. 34) was subject to assessment during Step 4 by Mechanical Engineering Assessors and subsequently a number of Assessment Findings were raised within the *Step 4 Mechanical Engineering Assessment of the EDF and AREVA UK EPR™ Reactor* (Ref. 37).

3.2 Identification of Representative Drop Load Cases from the Safety Class 1 Polar Crane in the Reactor Building, PEPS-G/2011/en/1060, Revision A.

49 The above submission (Ref. 17) was provided in response to the GDA Issue seeking a quantitative consequences analysis for Safety Class 1 lifting equipment and provided details of the representative cases that are to be subject to analysis, namely:

- Drop of the Reactor Pressure Vessel (RPV) closure head including drive shafts and lifting device – approximately 200 tonnes onto the RPV.
- Drop of a reactor cavity cover slab – approximately 70 tonnes onto the reactor cavity floor slab.
- Drop of a reactor cavity cover slab – approximately 70 tonnes onto the RPV closure head.
- Drop of the Multi-Stud Tensioning Machine (MSTM) including studs – approximately 80 tonnes onto the reactor cavity floor slab.

50 Information is provided within the submission relating to each of the representative lifts including the operational state of the reactor, details associated with the lifting rigs, locking mechanisms, lift lugs etc., and details of the load paths that are adopted for the lifts to be undertaken.

51 The submission details the potential scenarios that could occur as a result of a dropped load from the Polar Crane and includes details of the trajectory of the load, possible rotation and angle of descent and the potential source that is impacted.

3.3 Identification of Representative Drop Load Cases from the Safety Class 2 Cranes, ECEIG111781, Revision A.

52 The above submission (Ref. 18) was provided in response to the GDA Issue seeking a quantitative consequences analysis for Safety Class 2 lifting equipment and provided details of the representative cases that are to be subject to analysis, namely:

- Drop of a Fuel Assembly from the Refuelling Machine in the Fuel Pool in the Reactor Building.
- Drop of a Fuel Assembly from the Spent Fuel Mast Bridge in the Spent Fuel Pool of the Fuel Building.
- Drop of a Fuel Assembly from the Spent Fuel Mast Bridge in the Transfer Compartment of the Fuel Building.
- Container drop 20 feet long, 20 tonnes, from the Set Down Area Crane in the Fuel Building.

53 For each of the representative cases studies subject to analysis, there are details associated with the tools used, load paths and operating mode as well as specific details relating to the mass and dimensions of the dropped load. There are also details of the engineered systems in place e.g. limit switches and interlocks, as well as details of the operating modes of the lifting equipment.

3.4 Summary Report for the Substantiation of Dropped Loads Hazard, ECEIG120274, Revision A.

54 The above submission (Ref. 19) provides substantiation of the claims made relating to dropped loads and impact on civil structures within the UK EPR™. It provides the conclusions of the work undertaken in assessing dropped loads and impact from both Safety Class 1 and Safety Class 2 lifting equipment. The analyses undertaken consider the effects of dropped loads and impact from a civil engineering, mechanical, and safety case perspective.

55 The summary report details the outcome of the consequence assessments undertaken for the representative Safety Class 1 and Safety Class 2 lifting equipment as identified within Sections 3.2 and 3.3 above and identifies a number of design changes and operational measures to demonstrate that the consequences of a dropped load involving the lifting equipment on UK EPR™ are ALARP, namely:

- An increase to the shear reinforcement of the reactor cavity floor slab to [redacted]
- Manual connection of the Low Head Safety Injection/Residual Heat Removal (LHSI/RHR) system following a dropped load resulting in a Loss of Coolant Accident (LOCA).
- Removal of any of the three reactor cavity slabs above the RPV at refuelling boric acid concentration, at pressures less than 32 bar, and at temperatures less than 70^{o}C.

56 The civil engineering calculations undertaken as part of the consequence analysis were undertaken using a finite element analysis software tool which considered both bending and punching of the civil structures to assess the integrity of the reactor cavity slabs.

57 For Safety Class 2 lifting equipment, calculations were undertaken in accordance with the dropped loads civil methodology provided in response to **GI-UKEPR-IH-01.A2**.

58 An overview of the safety case presented for the representative Safety Class 1 and Safety Class 2 dropped loads is provided within Sections 3.4.1 and 3.4.2 below.

3.4.1 Dropped Loads and Impact Associated with Safety Class 1 Lifting Equipment

59 There are four postulated dropped loads considered for Safety Class 1 Lifting Equipment, their selection of which is justified within Reference 17. The Summary Report addresses each of the above events and makes reference to the supporting analysis that has been undertaken. An outline of the key aspects of the Summary Report together with reference to the supporting analyses is provided below.

60 In the case of drop of the RPV closure head, the analysis has been undertaken through strength of material calculations involving modelling using equivalent spring systems whose elasto-plastic stiffness is determined using either classic strength of material formulae or through 3D finite element models.

61 For drop of a reactor cavity cover slab onto the RPV closure head leading to a LOCA, the analysis of the consequences was modelled in two steps:

- Behaviour of the primary circuit and mass and energy release at the break are calculated using the primary circuit model in CATHARE v2.5 environment.
- Pressure and temperature evolution in the reactor building are then computed using the CONPATE 4 code.

62 The basis of the safety analysis undertaken comprises:

- The study of Safety Class 1 lifting equipment utilising the deterministic rules set for analysis of RRC events as defined for UK EPR™.
- The radiological acceptance criterion is set at the limits value of Plant Condition Category (PCC), PCC-4 events for the UK EPR™, namely, 10mSv.

63 The first case that is considered as part of the consequence analysis for the Safety Class 1 lifting equipment is drop of the RPV closure head onto the RPV. Figures 1 and 2 illustrate the lift of the head from the RPV together with an extract from the 3D model illustrating the location of the closure head in relation to the RPV.

[redacted]

[redacted]

[redacted]

[redacted]

64 The first case analysed is the postulated drop of the RPV closure head, which is considered to occur during outages when it is required to be lifted off the RPV to its' storage position to allow for maintenance and refuelling activities. The consequences of a drop of the RPV closure head from a height of 5 metres on to the RPV have been subject to analysis. The potential dropped load is assumed to fall directly on top of the RPV flange as the RPV closure head would still be engaged within guidance columns and as a result it would drop back down onto its initial position.

65 [redacted]. The submission states that there would be no damage of these components as they are capable of dissipating the entire kinetic energy of the postulated load drop and references the detailed consequence analysis, *"PEER-F DC 71 B, Consequences on the Reactor of an Accidental RPV Head Drop During it's Handling"* (Ref. 24) as the source of the justification. The report concludes that the risk to core cooling, residual heat removal and the risk of radiological releases are insignificant. No further reasonably practicable measures are identified within the ALARP analysis undertaken in Reference 23.

66 The second case analysed is associated with removal of the reactor cavity slabs and the potential consequences associated with drop of reactor cavity slab onto the slabs above the RPV. This operation is undertaken during an EPR outage at two reactor states depending on the reactor cavity slabs to be removed, namely:

- Reactor State B – Shutdown with secondary side heat removal utilising Steam Generators. At this state removal of the three reactor cavity cover slabs above the reactor cavity pool can take place.
- Reactor State C – Shutdown with primary side heat removal utilising the Residual Heat Removal System. At this state the three reactor cavity cover slabs above the RPV can be removed.

67 Figure 3, illustrates the load path for removal and subsequent storage position for the reactor cavity slabs.

[redacted]

[redacted]

68 The analysis undertaken considers a drop of a reactor cavity slab weighing approximately 70 tonnes from a height of 14 metres onto the reactor cavity floor slab. Given that the width of the slabs is wider than the opening to the reactor cavity, it is assumed that the slabs have tilted and impact on the floor slab at a number of different angles including point impact from the corner of one slab. Simulations have been undertaken using finite

element analysis within the supporting reference report, *"PECS-G/2011/en/0018 B, Check of Bearing Capacity of Reinforced Concrete Reactor Pool Slab Subject to Drop Load of a Concrete Cover Slab and a Multi-Stud Tensioning Machine."* (Ref. 26). The simulations conclude that there would be no perforation of the slab as well as no significant bending. As a result none of the fundamental safety functions of the UK EPR™ are threatened by a drop of the reactor cavity cover slab based upon assumed shear reinforcement of the reactor cavity floor slab. The following design change is identified associated with increasing the safety margin to an ALARP level:

- Increase to the shear reinforcement of the reactor cavity floor slab to [redacted]

69 The above design change has been captured within a formal Change Management Form (CMF), CMF34 (Ref. 11).

70 The report concludes that given the implementation of the design change coupled with maintaining the integrity of the reactor coolant pressure boundary and the reactor coolant system, the risk due to a reactor cavity cover slab drop from the polar crane on to the reactor cavity floor slab has been reduced to ALARP as presented within Reference 23.

71 The third case considers drop of a reactor cavity slab onto the RPV closure head. The removal of slabs is assumed to take place in Reactor State C2 with the Reactor Coolant System pressure below 32 bar with the temperature below 70^{o}C and the refuelling boron conditions met. The assumed consequences of a drop of a reactor cavity cover slab on the RPV closure head are:

- All 89 Control Rod Drive Mechanisms (CRDM) completely ruptured, resulting in a total break area of approximately [redacted]
- No fuel damage.

72 Figures 4 and 5, illustrate the RPV closure head including the location of the lifting rig and equipment on the head.

[redacted]

[redacted]

[redacted]

[redacted]

73 The analysis undertaken within the supporting reference, *"PEPR–F DC 85 B, Drop of a Reactor Cavity Cover Slab on the RPV Closure Head Analysis"* (Ref. 25) demonstrates that the controlled state is reached using automatic actions, with the LHSI/RHR system manually connected to reach a safe shut-down state. Further, the release of mass and energy associated with the impact results in temperatures and pressures remaining below the maximum permitted values in the PCC studies. Also, as the refuelling boron conditions are met there is no possibility of a return to criticality in the event of ejection of one or more Rod Cluster Control Assemblies (RCCA).

74 Reference 23 identifies the following two reasonably practicable operational measures to demonstrate the risk associated with a dropped cavity slab onto the RPV Closure Head is ALARP:

- Manual connection of the LHSI/RHR system following a dropped load resulting in a LOCA.
- Removal of any of the three reactor cavity slabs above the RPV at refuelling boric acid concentration, at pressures less than 32 bar, and at temperatures less than 70^{o}C.

75 The operational measures have been captured within a formal Change Management Form (CMF), CMF35 (Ref. 11).

76 The fourth representative case involved a drop of the Multi-Stud Tensioning Machine (MSTM) on to the reactor cavity floor slab. Figure 6 illustrates the location of the MSTM as well as the movements to and from the RPV cavity.

[redacted]

[redacted]

77 A number of load drops of the MSTM weighing approximately 80 tonnes from a height of 14m were simulated using finite element analysis. The civil engineering simulations undertaken in Reference 26 assumed a shear reinforcement of [redacted] and allowed for bending and punching. The civil engineering analysis concludes that perforation of the slab can be excluded and that no significant bending effects are observed.

78 As was the case for the analysis undertaken for the reactor cavity slabs (Ref. 25) there is a recommendation to increase the shear reinforcement to ensure the validity of the finite element analysis undertaken.

79 The summary report concludes that none of the fundamental safety functions of the UK EPR™ are threatened by a drop of the MSTM. Furthermore, it concludes that the reactor coolant pressure boundary is not be compromised nor are there any detrimental effects to cooling of the primary system.

3.4.2 Dropped Loads and Impact Associated with Safety Class 2 Lifting Equipment

80 There are four postulated dropped loads considered for Safety Class 2 Lifting Equipment, their selection of which is justified within Reference 18. The Summary Report addresses each of the above events and makes reference to the supporting analysis that has been undertaken. An outline of the key aspects of the Summary Report together with reference to any of the supporting analyses is provided below.

81 The first case analysed involves the drop of fuel assembly from the Refuelling Machine (RM) in the fuel pool in the Reactor Building. Figure 7 illustrates the Reactor Building Pool showing the locations of the Refuelling Machine (RM).

[redacted]

[redacted]

82 The representative case involves the drop of a fuel assembly from a height of 4m onto the thinnest part of the reactor cavity slab located in the vicinity of the Internals Storage section of the RB pool. The analysis within Reference 22 demonstrates that the fuel assembly would not perforate the slab due to it's thickness coupled with sufficient reinforcement to absorb the energy of impact by punching and bending.

83 The report concludes that a drop of a fuel assembly within the reactor cavity pool would not result in loss of more than one F1 redundancy as F1 components are not located within the potential impact volume of the RM as shown within the analysis undertaken within Reference 20. In addition, it does not result in PCC-3 or PCC-4 events other than a fuel handling incident within the RB whose consequences are bounded by the PCC-4 study within Chapter 14 of the PCSR (Ref. 12). The analysis considers that the measures in place are ALARP.

84 The second and third cases analysed are drops of a fuel assembly within the FB fuel pool. The second case considers a dropping a fuel assembly 6.5m from the Spent Fuel Mast Bridge (SFMB) into an area not protected by underwater fuel storage racks and the third considers a drop of a fuel assembly from a height of 5.28m in the FB Transfer Compartment.

85 Figure 8 illustrates the FB fuel pool together with locations for the SFMB over the Transfer Compartment and the Cask Loading Pit. In addition, it shows the location of the underwater fuel storage racks within the main part of the fuel pool.

[redacted]

86 As was the case for the RB, the analysis within Reference 22 demonstrates that the fuel assembly would not perforate the slab due to it's thickness coupled with sufficient reinforcement to absorb the energy of impact by punching and bending.

87 The summary report states that no more than one F1 redundancy is lost due to geographical separation of the F1 systems within the FB fuel pool and cites Sections 3.3.3 and 3.3.4 of Reference 20. As was the case for a drop a fuel assembly within the RB fuel pool, neither event results in PCC-3 or PCC-4 events other than a fuel handling incident within the FB whose consequences are bounded by the PCC-4 study within Chapter 14 of the PCSR (Ref. 12). The analysis considers that the measures in place are ALARP.

88 The fourth case analysed is the potential for a drop of a hypothetical container weighing 20 tonnes, which is the heaviest load handled by the Set Down Area Crane. The case that has been analysed is a flat drop of a container with an impact surface of 2.4m x 6.09m on a rectangular slab with dimensions 6m x 9m x 1m. The dimensions of the slab have been chosen in order to remain conservative. The crane is located within the Set Down Area of the FB and is generally only used during outages. Figure 9 illustrates the location of the crane within the Set Down Area.

[redacted]

[redacted]

89 Reference 22 is cited as providing the evidence that a drop of the container would not result in perforation of the slab and that the reinforcement is sufficient to absorb the energy of impact by punching and bending.

90 The report concludes that it is not possible to result in loss of more than one F1 redundancy given their location outside the potential impact volume and not being located within the Set Down Area. The evidence to support this is included within Sections 3.4.3 and 3.4.4 of Reference 20. In addition, no PCC-3 or PCC-4 events are generated and there are no radiological releases identified as a result a dropped load within the Set Down Area. The analysis considers that the measures in place are ALARP.

4 ONR ASSESSMENT

91 Further to the assessment work undertaken during Step 4 (Ref. 7), and the resulting GDA Issue **GI-UKEPR-IH-01** (Ref. 6), this assessment focuses on arguments and evidence identified within the EDF and AREVA deliverables. The deliverables are intended to provide the requisite evidence and were specified within the Resolution Plan (Ref. 8) provided by EDF and AREVA at the end of Step 4 of GDA.

92 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

93 The scope of the assessment has been to consider the expectations detailed within the GDA Issue, **GI-UKEPR-IH-01**, 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:

- Consequences of dropped loads and impact from lifting equipment within the UK EPR™ design including the consideration of civil structures, additional physical protection, limits and conditions of use of the lifting equipment, load paths, and administrative controls.
- Details of the approach taken to treat dropped loads on civil structures.

94 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

95 The following deliverables submitted in response to the GDA Issue have been subject to detailed assessment as part of the GDA close out:

- Dropped Loads – Summary of Design Basis and Principles (Ref. 16).
- Identification of Representative Drop Load Cases from the Safety Class 1 Polar Crane in the Reactor Building (Ref. 17).
- Identification of Representative Drop Load Cases from the Safety Class 2 Cranes (Ref. 18).
- Summary Report for the Substantiation of Dropped Loads Hazard (Ref. 19).
- Methods with regard to the risk of dropped loads for UK EPR™ for concrete structure (Ref. 27).

96 It is important to note that the submission, "Summary Report for the Substantiation of Dropped Loads" provides a summary of the work that has been undertaken to underpin the safety case for dropped loads and impact. Given the number of references provided, the summary report has been assessed in detail and the references sampled in accordance with the ONR HOW2 guidance PI/FWD.

97 As mentioned previously the assessment of the submission, "Methods with regard to the risk of dropped loads for UK EPR™ for concrete structure" is not detailed within this internal hazards assessment; however, the conclusions arising from the assessment are provided together with a reference to the civil engineering assessment.

4.2.1 Dropped Loads – Summary of Design Basis and Principles, ECEIG111683 Revision A

98 As the claims, arguments, and evidence for dropped loads are primarily associated with the design with supplementary supporting arguments and evidence associated with the administrative controls applied to the crane, it is this design aspect of the submission that this assessment focuses upon on a sampling basis.

99 The standards applied to the design of the lifting equipment for UK EPR™ are the Nuclear Safety Standards Commission (KTA) standard, KTA 3902, “Design of Lifting Equipment in Nuclear Power Plants” (Ref. 31) and the EDF standard, BTS 60.C.007.03, “High Safety Lifting and Handling Machines (Ref. 32). These standards identify the single failure proof nature of the lifting equipment, as is the case with existing NUREG standards, NUREG 0554 (Ref. 35) and NUREG 0612 (Ref. 36). Discussions have taken place with mechanical engineering specialists and they have confirmed that the design standards applied are consistent with those applied worldwide and they consider that they are established and robust. The mechanical engineering specialists judge that the application of a consequence analysis is consistent with our SAPs and would expect that the representative cases be analysed for Safety Class 1 and Safety Class 2 lifting equipment. This is based upon the operational experience associated with dropped loads and impact not being associated with the design, but more to do with human performance aspects. In addition, the mechanical engineering specialists stated that the standards adopted for the design do not make the potential for dropped loads and impact incredible as there is the need to consider a number of other aspects including design, procurement, installation, and operational requirements. They also stressed the importance that, although the codes seem reasonable, they are prescriptive and there would be a need to consider whether there are any further ALARP measures.

100 It is positive to note that EDF and AREVA has undertaken a review of operating feedback for dropped loads and impact that have occurred worldwide. The information from the review provides confidence in the design standards of the lifting equipment for Safety Class 1 and Safety Class 2. The review identified that even though the lifting equipment is to a high standard of design, dropped loads and impacts cannot be ruled out due to other factors involving human performance. This therefore, supports my judgement during Step 4 that quantitative consequence analysis is required to provide confidence that should there be a failure in the administrative controls in place that the risk is either ALARP or further mitigation measures are necessary. These further measures may be in the form of further engineered protection systems, enhanced administrative control, or prevention of specific lifts during certain operational states. The representative lifts from the Safety Class 1 and Safety Class 2 lifts have been identified within Section 3.2 and 3.3 respectively and the associated assessment has been undertaken within Sections 4.2.2 and 4.2.3.

101 As mentioned previously, *“UK EPR GDA – Management of Nuclear Safety Significant Lifting”* (Ref. 34) has been subject to assessment by mechanical engineering specialists and documented within the Step 4 Mechanical Engineering Assessment of the EDF and AREVA UK EPR™ (Ref. 37). The assessment concluded that, subject to resolution of the Assessment Findings, **(AF-UKEPR-ME-14, 15, 16, 17, and 18)**, they were satisfied with the justification provided in respect of nuclear lifting and design principles for the UK EPR™ from a GDA perspective against SAP EDR.1. Clarification was sought from the mechanical engineering assessor who undertook the assessment to determine whether his assessment had addressed any aspects of quantitative consequence analysis. Confirmation was provided which stated that the review had been limited to the mitigation in place to minimise the potential for dropped loads and impact accounting for load path

and rigging arrangements, engineered provisions such as interlocks and end stops, and aspects of administrative control. As these aspects dealt with primarily the design and existing arrangements for control of lifting, there is to be no further assessment of the submission as part of this assessment.

102 Overall, I conclude that the submission provided detailing the summary of the design basis forms a comprehensive approach to design, operational experience, and that the submission identifies the need to undertake a quantitative consequence analysis as part of the overall case for dropped load and impact hazards in the frame of the GDA.

4.2.2 Identification of Representative Drop Load Cases from the Safety Class 1 Polar Crane in the Reactor Building, PEPS-G/2011/en/1060, Revision A.

103 During Step 4 it was identified that EDF and AREVA proposed to operate the Polar Crane within Containment during operational states when the reactor could be at temperatures greater than 120^{o}C and pressures less than 130 bar. The approach currently undertaken within the UK for the analysis of dropped loads associated with the lifting equipment involves the assessment of the consequences of dropped loads on safety significant SSCs which results in the determination of the limits and conditions of operation of the lifting equipment, detailed load paths, and systems and administrative controls in place. In addition, current practice employed at the existing UK PWR and within other plants internationally is for the reactor to achieve cold shutdown, with temperatures <93 ^{o}C and pressures <30 bar, prior to undertaking operations involving the Polar Crane.

104 As a result, the GDA Issue required EDF and AREVA to produce a quantitative consequence analysis for lifts involving Safety Class 1 lifting equipment. It was recognised that not all lifts of such equipment would have been developed for the UK EPR™ at this stage. It was agreed that the analysis would involve four representative lifts (consisting of five dropped loads) at differing operational states to provide confidence that the lifts would be bounding.

105 It is accepted that the representative cases proposed by EDF and AREVA will provide a high degree of confidence that the consequences of a dropped load or impact within the area will be bounding. The approach taken to the initial qualitative analysis of dropped loads within the submission is positive.

106 Overall the submission provides the requisite information relating to the identification of the potential dropped load and impact scenarios in order to support the quantitative consequence analysis, further details of which are discussed within Section 4.2.4.

4.2.3 Identification of Representative Drop Load Cases from the Safety Class 2 Cranes, ECEIG111791, Revision A.

107 The basis for requiring quantitative consequence assessment associated with Safety Class 2 cranes was due to the implicit claims made upon the civil structure in the event of a dropped load. It was not clear to ONR how structures were claimed within the safety case for this purpose given the number and variance in the lifts to be undertaken by such lifting equipment. Again, it was requested that a number of representative lifts be identified and for the analysis to consider the most onerous or bounding conditions with the current known lifts to provide confidence that the claims made upon the civil structures were valid.

108 The submission identifies that Safety Class 2 cranes within the UK EPR™ design are mainly related to Fuel Handling within the Reactor Building and Fuel Building. As was the

case for Safety Class 1 cranes, a qualitative assessment undertaken relating to the potential dropped loads and impact has been undertaken.

109 In the case of a drop of a Fuel Assembly from the Refuelling Machine in the Reactor Building, there is information provided relating to the Fuel Assembly Gripper which is attached to the Refuelling Machine and has a number of guidance pins and gripping fingers to ensure that the Fuel Assembly is secured to the Fuel Assembly Gripper prior to movement. In addition, there are details relating to the load path and operating mode of the Refuelling Machine when undertaking this lift. The information provided within the report gives confidence that the engineered Fuel Assembly Gripper coupled with the load path and operating analysis should demonstrate the potential for a dropped load from the Fuel Assembly is low. However, the quantitative consequence analysis provided as part of the final submission for dropped loads and impacts has been produced to provide confidence that the risk to nuclear safety arising from a dropped fuel assembly is ALARP. A similar approach has been adopted for the other lifts involving Fuel Assemblies within the Fuel Building. Once again, there is detailed information relating to load paths and operating modes which considers the areas of the Spent Fuel Pool in which the Fuel Assemblies can travel.

110 The lifting equipment is controlled using a Programmable Logic Controller (PLC) which limits the travel, height and utilises interlocks to prevent dangerous movements of the lifting crane. Given the deterministic analysis of the consequences of dropped loads and impact I have not considered the safety case claims, if any, placed upon these control systems within my assessment.

111 As was the case for Safety Class 1 lifting equipment, the representative cases for Safety Class 2 lifting equipment proposed by EDF and AREVA will provide a high degree of confidence that the consequences of a dropped load or impact within the area will be bounding. The approach taken to the initial qualitative analysis of dropped loads within the submission is positive.

112 Overall the submission provides the requisite information relating to the identification of the potential dropped load and impact scenarios in order to support the quantitative consequence analysis, further details of which are discussed within Section 4.2.4.

4.2.4 Summary Report for the Substantiation of Dropped Loads Hazard, ECEIG120274, Revision A.

113 The above submission provides substantiation of the claims made relating to dropped loads and impact on civil structures within the UK EPR™. It provides the conclusions of the work undertaken in assessing dropped loads and impact from both Safety Class 1 and Safety Class 2 lifting equipment. The analyses undertaken consider the effects of dropped loads and impact from a civil engineering, mechanical, and safety case perspective.

114 The report details the outcome of the postulated dropped loads detailed within Sections 3.2 and 3.3, which are subjected to assessment within Sections 4.2.2 and 4.2.3 of this report. The summary report identifies a number of design changes and operational measures to demonstrate that the consequences of a dropped load involving the lifting equipment on UK EPR™ are ALARP.

115 For each of the postulated load drops, the submission, *“ALARP Justification of Identified Representative Drop Load Cases from the Safety Class 1 Polar Crane in the Reactor Building”* (Ref. 23) is referenced, which provides further details of the design measures in place including the control and protection systems and operational control measures. It is

explained that the Polar Crane is manually operated by Suitably Qualified and Experienced Person (SQEP). The control of movements is by sight from either the main control desk or a portable box with a Programmable Logic Controller (PLC) that monitors, alarms, and limits movements through the provision of limit switches and mechanical interlocks. I am satisfied with the approach taken to the control of movements through the use of both engineered protection and operational controls as they are in line with current expectations and relevant good practice within the UK. However, the Assessment Findings produced as a result of the Step 4 Mechanical Engineering Assessment of the EDF and AREVA UK EPR™ Reactor (Ref. 37) **(AF-UKEPR-ME-14 – 18)** should provide the evidence associated with rigging and load paths for the UK EPR™.

116 My assessment focuses on the detailed analysis undertaken by EDF and AREVA to underpin the dropped loads and impact cases identified for Safety Class 1 and Safety Class 2 lifting equipment. Each of the postulated load drop events have been subject to assessment within this section including assessment of the evidence cited in support of the conclusions drawn.

4.2.4.1 Drop of the Reactor Pressure Vessel (RPV) closure head including drive shafts and lifting device – approximately 200 tonnes onto the RPV.

117 The assumptions associated with the drop of the RPV closure head are reasonable given that the head would still be in the guidance columns and any drop would involve a drop directly onto the top of the RPV into its initial position. When the RPV head is lifted above the guidance columns it is assumed that it would topple and impact on the reactor pool walls, which appears reasonable, however, the detailed consequence analysis undertaken (Ref. 24) has been sampled as part of my assessment to determine the acceptability of the evidence presented.

118 The analysis presented considers two scenarios:

- RPV closure head drops onto the RPV flange prior to disengaging the vessel head guidance columns. The potential dropped load in this case being 5 metres.
- RPV closure head drops onto vessel internals once it has disengaged the vessel head guidance columns. The drop is postulated to occur between the RPV and the service floor with a maximum potential dropped load height of 18 metres.

119 In the first case, the analysis provides detailed calculations associated with the kinetic energy involved and the resultant force on the RPV flange. The components of the RPV considered within the analysis are:

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

120 The four components are assumed to be impacted by the drop of the RPV closure head and in all cases, it is assumed that the head drops in water. This is a reasonable assumption given that the pool would be flooded up as the head was removed.

121 The calculations undertaken for the four components make conservative assumptions that involve considering 100% of the impact energy analysed and assuming that surrounding structures are rigid and therefore do not absorb any of the kinetic energy generated by the event. In addition the drag coefficient applied is equal to 1, which is taken to be

conservative given that RPV head would have a higher drag coefficient as it passes through the water.

122 [redacted]

123 [redacted]

124 I am satisfied that the approach and method of calculating the impact energy on the components identified is acceptable and is derived from straightforward calculation assuming conservatism in the speed of the RPV head through water.

125 The impact energy is then considered on the 4 steel components identified above using their respective material properties and it is shown that in each case a drop of the RPV closure head would not result in failure of the component in question. A limited review of the analysis has been undertaken and found that, based upon the comprehensive analysis undertaken utilising finite element analysis together with consideration of the margins to failure, the conclusions are reasonable.

126 I am, therefore, satisfied that the dropped load and impact of the RPV closure head for this postulated scenario has been subject to detailed quantitative consequence analysis and has adequately demonstrated that the provisions for dropped loads in this case are ALARP.

127 In the case of the RPV closure head dropping onto vessel internals once it has disengaged the vessel head guidance columns, the consequence analysis provides evidence to support the acceptability of such a dropped load.

128 Figures 10 and 11 illustrate the load path of the RPV closure head as well as the potential location postulated for the dropped load.

[redacted]

[redacted]

[redacted]

[redacted]

129 There is both qualitative and quantitative evidence provided for this case. The assumption of the location of the drop in between the RPV and the Service Floor would impact on the reactor pool walls is entirely reasonable given the available space around the flange of the RPV. The water coverage would also serve to slow down the descent and ultimately reduce the impact of the RPV closure head on the vessel internals. I am

satisfied that the potential for a drop of 18 metres on to this area would be very unlikely to fall in such a way that it lined up with the guidance columns and hence impact squarely onto the RPV flange. However, a consequence analysis has been undertaken which considers this event and shows that there would be no structural damage to the fuel assemblies and the impact would be within their designed withstand capability.

130 The detailed calculations associated with this event have not been subject to sampling assessment within this report given the likelihood of the event based upon the physical geometry of the RPV and the RPV closure head.

131 I am satisfied that a drop of the RPV closure head for the above event associated with an 18 metre drop onto the RPV has been subject to detailed quantitative consequences analysis which demonstrates that the provisions in place are ALARP.

4.2.4.2 Drop of a reactor cavity cover slab – approximately 70 tonnes onto the reactor cavity floor slab.

132 The potential for a drop of a reactor cavity cover slab considers two reactor states; Reactor State B and Reactor State C. This scenario is associated with a drop of one of the three rector cavity cover slabs above the reactor cavity pool. The approach to undertaking the lift during Reactor State B is acceptable as there is no potential for a drop of a slab to directly impact onto the RPV closure head.

133 Figure 12 provides an illustration of the location of the slabs on the supports of the reactor cavity pool together with the locating lugs. It should be noted that Figure 12 is not represented correctly as it is not possible to remove the reactor cavity cover slabs before those over the Fuel Pool. Figure 13 illustrates a section through the Containment which shows how the slabs would be removed.

[redacted]

[redacted]

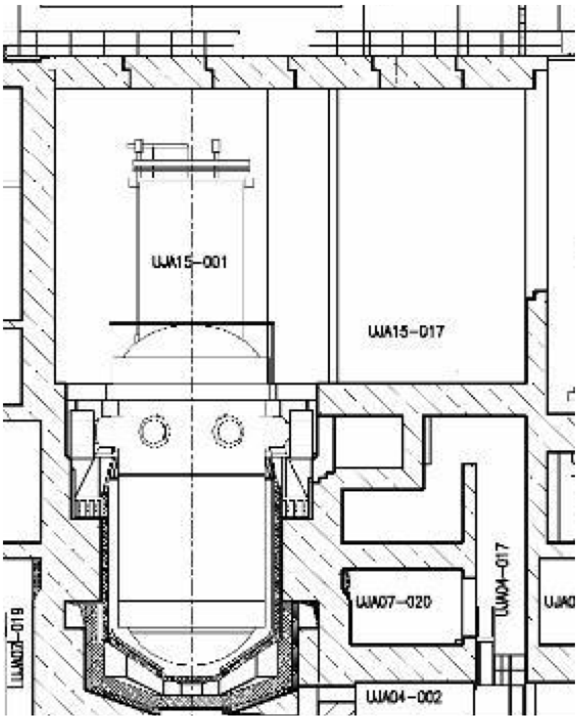

Figure 13: Section through the Containment illustrating the removal sequence for the Reactor Cavity Cover Slabs located above the Reactor Cavity Pool

134 The analysis detailed within Reference 26 has considered the consequences of a dropped reactor cavity cover slab onto the reactor cavity floor slab and reported the outcome of the finite element analysis. The finite element analysis has not been subject to assessment as the calculations are associated with the civil engineering design of the structures which was subject to assessment during Step 4 of GDA by civil engineering assessors. The *"Step 4 Civil Engineering and External Hazards Assessment of the EDF and AREVA UK EPR™ Reactor"* (Ref. 38) identified an Assessment Finding that required the licensee to take account of any implications of the outcomes of the Internal Hazards GDA Issues which could affect the design of civil structures **(AF-UKEPR-CE-05)**. The timescale for the finding is that it should be completed ahead of the placement of first structural concrete to ensure that there are no options foreclosed for mitigation of internal hazards as a result of concrete placed.

135 The following modifications (Ref. 11) were issued to ONR during the GDA close out for inclusion within the design reference (Ref. 9):

- CMF34, which relates to increasing the shear strength of the reactor cavity floor slab.
- CMF35, which relates to the ability to switch the LHSI pumps from RHRS mode to injection mode as well as specifies that the reactor cavity cover slabs cannot be removed until Reactor State C is reached, at least [redacted] after reactor trip.

136 CMF34 states that the design change is required further to the analysis and I am satisfied that the CMF be included within the design reference (Ref. (9). CMF35 does identify further analysis work to be undertaken as part of the development of the modification, however, I am content that it can be included within the design reference (Ref. 9) given that the modification has been identified as being required.

137 From an internal hazards perspective, I am satisfied with the load drop scenario as well as the operational requirements associated with the removal of the three reactor cavity cover slabs.

4.2.4.3 Drop of a reactor cavity cover slab – approximately 70 tonnes onto the RPV closure head.

138 This scenario is associated with a drop of one of the reactor cavity cover slabs above the RPV at Reactor State C. During Step 4 I raised concerns over the potential for lifting the reactor cavity cover slabs at any temperature and at pressures greater than 130 bar. The submission provided identifies a design change that now prevents lifting of the any of the three reactor cavity cover slabs that are above the RPV unless the reactor is at refuelling boric concentration and temperatures less than 70^{o}C. This is undertaken at Reactor State C2 with the RCS pressure below 32 bar and with the time since reactor trip greater than [redacted]. This is a positive improvement to the approach to mitigating the consequences of a dropped load involving the reactor cavity cover slab.

139 Reference 25 provides a detailed analysis of the potential consequences of a drop of the reactor cavity cover slab at Reactor State C.

140 Within Reference 25, it is claimed that the slab cannot fall directly onto the RPV closure head without being tilted due to them being wider than the cavity width [redacted] and as such they would need to tilt and/or rotate in order to fall into the cavity and impact on the closure head. The consequence analysis pessimistically assumes that this would result in loss of the all the 89 Control Rod Drive Mechanisms (CRDM) with a resultant break in the primary circuit of [redacted]. Given the physical limitations associated with

the reactor cavity cover slab and the assumed loss of all CRDMs, I am satisfied that these assumptions are conservative.

141 There is a claim that the RPV closure head would not fail and result in a more significant release. The evidence to support this is that the slab would not be able to impact the RPV closure head due to the damping effect of the equipment above coupled with the thickness of the slab [redacted] limiting the potential for it to reach the head. I am satisfied that the drop of a reactor cavity cover slab would not impact the RPV head based upon the evidence provided within the analysis.

142 The analysis then provides the sequence of events involving the requirements for decay heat removal and alignment of the LHSI/RHR systems. The approach taken to the analysis of the sequence associated with the safety system requirements has been comprehensive and it has identified the need for operator action to ensure decay heat removal through a manual connection of the LHSI/RHR following a LOCA arising from a dropped load from the Polar Crane. This design change, CMF35 (Ref. 11) appears reasonable given the need to attain a safe shutdown state without heat removal means with the RCS or within Containment. The analysis identifies the need for the operator action to be subject to further studies relating to both the system and operational aspects. I welcome the need to consider the impact of this modification in greater detail and given the need to achieve a safe shutdown state, I have raised the following Assessment Finding to ensure that it is captured during the Site Specific Phase:

> ***AF-UKEPR-IH-9:*** *The Licensee shall ensure that the further studies in order to support the design modification associated with the manual connection of the LHSI/RHR system are appropriately considered within the site specific design.*
>
> ***Required Timescale:*** *Mechanical, Electrical, and C&I Safety Systems – Before inactive commissioning.*

143 I am satisfied that the detailed consequences assessment adequately addresses the potential nuclear safety impact associated with the drop of a reactor cavity cover slab onto the RPV for the Reactor State analysed. The outcome of the further studies with regard to the operator actions will need to demonstrate the totality of the provisions in place to protect against the event are ALARP, **AF-UKEPR-IH-9** refers.

4.2.4.4 Drop of the Multi-Stud Tensioning Machine (MSTM) including studs – approximately 80 tonnes onto the reactor cavity floor slab.

144 The evidence presented associated with the consequences of a drop of the MSTM onto the reactor cavity floor slab is the same as that presented for drop of a reactor cavity cover slab within the pool.

145 I am satisfied from an internal hazards perspective, with the load drop scenario, as well as the operational requirements associated with the load paths defined. I am, therefore, satisfied with the analysis and evidence presented associated with a drop of the MSTM onto the reactor cavity floor slab.

4.2.4.5 Dropped Load Scenarios identified from Safety Class 2 Lifting Equipment

146 For the scenarios associated with dropped loads from Safety Class 2 lifting equipment, the claims are predominantly associated with the civil structures onto which the dropped load impacts. The Summary Report (Ref. 19) provides the claims on the structures and refers out to the submission, *"Safety Case for 4 Representative Load Drops from Safety "Classified 2 Cranes"* (Ref. 20) and *"EPR UK – RS2 cranes – Drop Load Impact*

Calculations" (Ref. 22) as the source of the detailed safety analysis undertaken for Safety Class 2 lifting equipment.

147 I have therefore elected to subject References 20 and 22 to further assessment. I have chosen to assess the drop of a fuel assembly in the Fuel Pool of the Reactor Building as part of my sample as the evidence presented for each of the potential dropped load scenarios for Safety Class 2 lifting equipment is very similar:

148 Reference 20 provides details of the basis of the designation of a Safety Class 2 piece of lifting equipment from a safety perspective, which require it to be Safety Class 2 if the consequences of a dropped load could lead to:

- A non-isolatable release of primary coolant into the Containment, or,
- A failure which leads to consequential failure of an F1 system, or,
- A release of radioactivity leading to increased radiation levels inside the area which affects the classification of radiological zones.

149 The approach to the need to designate lifting equipment on this basis is an acceptable method by which to differentiate between the requirements of Safety Class 1, Safety Class 2, and non-safety lifting equipment.

150 The safety analysis considers the following deterministic rules when considering dropped loads and impact:

- A dropped load is postulated only for one item of equipment at a time.
- The dropped load occurs during normal plant operating conditions (power operation or shutdown conditions.
- A dropped load may occur simultaneously with a facility fault, or when plant is unavailable due to maintenance.
- There is a significant potential for hazards to act as initiators of common cause failure, including loss of off-site power and other services.
- Dropped loads have the potential to threaten more than one level of defence in depth at any one time.
- Dropped loads can arise as a consequence of events external to the site and should be included in the relevant fault sequences.

151 I am satisfied with the basis and rules associated with the deterministic approach to dropped loads as it is line with current UK expectations.

152 References 20 and 22 cite the methodology applied to load drops on concrete structures (Ref. 27) which has been subject to assessment by civil engineering assessors (Ref. 39), the conclusions of which are provided in response to GDA Issue Action, **GI-UKEPR-IH-02.A2**, within Section 4.2.5 of this assessment report.

153 In addition to the claims made on the civil structures, the drop of a fuel assembly from the Refuelling Machine (RM) in the Refuelling Pool of the Reactor Building provides arguments associated with:

- Design and procurement of the lifting equipment.
- Testing and maintenance.
- Operating instructions.

154 Discussion of the design and procurement of the lifting equipment is considered within Section 4.2.1 of this assessment report, however, further information associated with the main design provisions in place is provided. The design provisions include requirements for the Fuel Assembly gripper for both normal and seismic conditions taking into account the maximum load that the gripper would be expected to handle. There are also limit switches and interlocks included within the design to prevent dangerous movements to prevent load interactions. In addition, there are design provisions in place to prevent overspeed when the RM is within the predefined circulation area as well as systems in place to ensure that the RM decelerates when approaching specific locations. Finally, there are systems in place for load monitoring and the application of brakes and, in the event of failure of the sensors, the PLC cuts power the supply to the motors and actuates the brakes.

155 The submission provides limited information relating to testing and maintenance of the lifting equipment other than to state that commissioning tests will be undertaken. I am satisfied that these aspects are addressed at the Site Specific Phase as these are not considered to impact on the PCSR for GDA. I have not raised an Assessment Finding as this is considered to be part of the licensee’s process and I do not have concerns relating to the adequacy or need to capture as an AF.

156 The operating instructions are driven by the need to identify safe load paths and identify potential rigging faults, both aspects of which have been captured as AFs within the Step 4 Mechanical Engineering Assessment of the EDF and AREVA UK EPR™ Reactor (Ref. 37) as mentioned previously in my assessment report.

157 I am satisfied with the approach taken to the design and control of movements for Safety Class 2 lifting equipment through the use of both engineered protection and operational controls as they are in line with current expectations and relevant good practice within the UK.

4.2.5 Methods with regard to the risk of dropped loads for UK EPR for concrete structure, ENGSGC100483, Revision A.

158 As mentioned previously the evidence associated with dropped loads and impact for Safety Class 2 lifting equipment is predominantly associated with civil engineering calculations undertaken within the above submission (Ref. 27), which was provided in response to **GI-UKEPR-IH-02.A2**. Given that the report provides detailed civil engineering calculations, the assessment of the submission was undertaken by civil engineering assessors and reported within Reference 39.

159 The civil engineering assessment concluded that they were satisfied with the dropped load methodology document (Ref. 27) as it includes a sufficient range of methods which are appropriate for the types of dropped loads that could occur within a nuclear power plant. However, the selection of the actual method to be used for each dropped load scenario awaits the characterisation of the dropped load concerned. Two Assessment Findings **(AF-UKEPR-CE-81 and AF-UKEPR-CE-82)** have been raised within reference 39 associated with the civil engineering methodology for dropped loads and impact.

4.3 Comparison with Standards, Guidance and Relevant Good Practice

160 In terms of internationally accepted standards and guidance, operating experience and relevant good practice, it was considered important to provide an overview of the current expectations associated with dropped loads and impact from both a national and international perspective.

161 The HSE Safety Assessment Principles, SAPs, state within EHA.14:

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.		

162 The approach currently undertaken within the UK for the analysis of dropped loads associated with the lifting equipment involves the assessment of the consequences of dropped loads on safety significant SSCs which results in the determination of the limits and conditions of operation of the lifting equipment, detailed load paths, and systems and administrative controls in place. In addition, current practice employed at the existing UK PWR and within other plants internationally is for the reactor to achieve cold shutdown, with temperatures <93 degrees Celsius and pressures <30 bar, prior to undertaking operations involving the Polar Crane. As a result of the consequence analysis that has been undertaken, EDF and AREVA have now proposed a design change that now prevents lifting of any of the three reactor cavity cover slabs above the RPV unless the reactor is at refuelling boric concentration and temperatures less than 70^{o}C. This is undertaken at Reactor State C2 with the RCS pressure below 32 bar and with the time since reactor trip greater than [redacted]. This is in line with UK expectations and in line with other plants internationally.

163 NS-G-1.11 (Ref. 5) states, "Structures classified as liable to affect SSCs in the event of their collapse should be designed and built so that the probability of their collapsing can be shown to be negligible; otherwise the consequences of their collapse should be evaluated. Similarly, the hazard posed to SSCs by falling objects (cranes and lifted loads) should be evaluated". The approach to the analysis of the consequences within NS-G-1.11 is consistent with the approach adopted within the UK currently and UK EPR™ is consistent with the expectations associated with the need to perform consequence analysis.

164 In addition to NUREG-0554 (Ref. 35), the USNRC issued NUREG-0612 (Ref. 36), which presented an overall philosophy that provided a defence-in-depth approach for controlling the handling of heavy loads with the focus on prevention of dropped loads rather than assessment of the consequences and it subsequently required the following approach to be adopted within existing US Nuclear Power Plant:

- Assure that there is a well designed handling system.
- Provide sufficient operator training, load handling instructions, and equipment inspection to assure reliable operation of the handling system.
- Define safe load travel paths and procedures and operator training to assure to the extent practical that heavy loads are not carried over or near irradiated fuel or safe shutdown equipment.
- Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.
- Where mechanical stops or electrical interlocks cannot be provided provide a single-failure-proof crane or perform load drop analyses to demonstrate that unacceptable consequences will not result.

165 The current design for UK EPR™ appears to be consistent with the philosophy stated within NUREG 0612 and through the production of detailed consequence analysis

including load path and detailed impact calculations, together with additional electrical and mechanical interlocks, have demonstrated the requisite arguments and evidence to support this.

5 REVIEW OF THE UPDATE TO THE PCSR

5.1 Internal Hazards

166 Section 5 of Chapter 13.2 of the PCSR update (Ref. 40) considers dropped loads and impact. The submission was reviewed to ensure that the outcome of the GDA assessment and subsequent design changes had been appropriately captured therein.

167 The following design changes, CMFs 34 and 35 (Ref. 11), were identified within the references to the revised PCSR as a result of the GDA Issue associated with dropped loads and impact:

- An increase to the shear reinforcement of the reactor cavity floor slab to [redacted]
- Manual connection of the Low Head Safety Injection/Residual Heat Removal (LHSI/RHR) system following a dropped load resulting in a Loss of Coolant Accident (LOCA).
- Removal of any of the three reactor cavity slabs above the RPV at refuelling boric acid concentration, at pressures less than 32 bar, and at temperatures less than 70^{o}C.

168 The PCSR has been reviewed and I am satisfied that it reflects the findings from the GDA and the text has been updated to include reference to the supporting analysis work undertaken within References 20, 23, and 34. In addition, an overview of the detailed consequence analysis is provided for both the Safety Class 1 and Safety Class 2 lifting equipment.

6 ASSESSMENT FINDINGS

6.1 Additional Assessment Findings

169 The following Assessment Finding has been raised that are required to be resolved during the site specific phase:

> ***AF-UKEPR-IH-9:*** *The Licensee shall ensure that the further studies in order to support the design modification associated with the manual connection of the LHSI/RHR system are appropriately considered within the site specific design.*
>
> ***Required Timescale:*** *Mechanical, Electrical, and C&I Safety Systems – Before inactive commissioning.*

6.1.1 Impacted Step 4 Assessment Findings

170 No Assessment Findings raised during Step 4 have been impacted as a result of this assessment.

7 ASSESSMENT CONCLUSIONS

171 Further to the receipt of the deliverables detailed within the Resolution Plan comprising of quantitative consequence analyses undertaken for dropped loads and impact arising from Safety Class 1 and Safety Class 2 lifting equipment, I am satisfied that the safety case for dropped loads and impact for the UK EPR™ is adequate.

172 My judgement is based upon the following factors:

- The approach to analyse the quantitative consequences of dropped loads and impact for Safety Class 1 and Safety Class 2 lifting equipment is in line with the HSE SAPs as well as internal guidance and relevant good practice.
- The analyses provided are comprehensive and have found that the consequences of a dropped load or impact from lifting equipment proposed for the UK EPR™ are acceptable to nuclear safety.
- The design of the lifting equipment is to a high standard and consistent with expectations within the United Kingdom and worldwide.
- The approach to the analysis of the consequences of failure, together with the operating conditions, is in line with the expectations of mechanical engineering assessors within ONR.
- The claims made associated with the civil structures have been subject to assessment by civil engineering assessors and found to be acceptable.
- EDF and AREVA have identified design changes as a result of the consequence analyses undertaken which has demonstrated that the provisions in place to protect against a dropped load or impact associated with Safety Class 1 and Safety Class 2 lifting equipment are ALARP.

173 One Assessment Finding has been raised in relation to this assessment, which requires a future Licensee to ensure that the further studies in order to support the design modification for the manual connection of the Low Head Safety Injection/Residual Heat Removal (LHSI/RHR) system are appropriately considered within the site specific design.

174 The Stage 2 Change Modification Forms (CMFs) associated with dropped loads and impact have been submitted to ONR (Ref. 11). The two CMFs, CMF34 and CMF35, have been reviewed and as they capture the design changes identified as a result of the analyses undertaken in response to this GDA Issue, I am satisfied that they can be included within the design reference (Ref. 9). In addition, the design changes have been captured within the PCSR update (Ref. 40).

175 The updated PCSR (Ref. 40) has been reviewed and I am satisfied that the outcome of the analyses undertaken has been adequately reflected therein.

176 I am, therefore, satisfied that GDA Issue, **GI-UKEPR-IH-01**, 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.

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

Nuclear Lifting Operations. T/AST/056 Issue 02, 01 December 2011

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

4 *Not used.*

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.

www.iaea.org.

6 *GDA Issue GI-UKEPR-IH-01 Revision 2.* ONR. July 2011. TRIM Ref. 2011/362813. (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-01 Revision 2.* EDF and AREVA. June 2011. TRIM Ref. 2011/302579. (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 *Deliverable to GI-UKEPR-CC-02 Action 1 – Design Change Submission Programme Revision 16,* Letter from EDF and AREVA to ONR, EPR01322R, 24 August 2012. TRIM Ref. 2012/337759.

12 *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.

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

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

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

16 *Dropped Loads – Summary of Design Basis and Principles.* ECEIG111683 Revision A, EDF, September 2011. TRIM Ref. 2011/496942.

17 *Identification of Representative Drop Load Cases from the Safety Class 1 Polar Crane in the Reactor Building.* PEPS-G/2011/en/1060 Revision A, AREVA NP, September 2011. TRIM Ref. 2011/513725.

18 *Identification of Representative Drop Load Cases from the Safety Class 2 Cranes.* ECEIG111791 Revision A, EDF, October 2011. TRIM Ref. 2011/513710.

19 *Summary Report for the Substantiation of "Dropped Loads" Hazard.* ECEIG120274 Revision A, EDF, February 2012. TRIM Ref. 2012/116596.

20 *Safety Case for 4 Representative Load Drops from Safety Classified 2 Cranes.* ECEIG120198 Revision A, EDF, February 2012. TRIM Ref. 2012/116594.

21 *Application Note for a Drop Load Impact on a Reinforced Concrete Slab.* ECEIG111395 Revision A, EDF, March 2012. TRIM Ref. 2012/116591.

22 *EPR UK – RS2 cranes – Drop Load Impact Calculations.* ECEIG111620 Revision A, EDF, February 2012. TRIM 2012/116593.

23 *ALARP justification of Identified Representative Drop Load Cases from the Safety Class 1 Polar Crane in the Reactor Building.* PEPS-G/2011/en/1076 Revision C, AREVA NP, February 2012. TRIM 2012/116601.

24 *Consequences on the Reactor of an Accidental RPV Head Drop During it's Handling.* PEER-F DC 71 Revision B, AREVA NP, February 2012. TRIM Ref. 2012/116598.

25 *Drop of a Reactor Cavity Cover Slab on the RPV Closure Head Analysis.* PEPR–F DC 85 Revision B, AREVA NP, February 2012. TRIM Ref. 2012/116599.

26 *Check of Bearing Capacity of Reinforced Concrete Reactor Pool Slab Subject to Drop Load of a Concrete Cover Slab and a Multi-Stud Tensioning Machine.* PECS-G/2011/en/0018, Revision B, AREVA NP, February 2012. TRIM Ref. 2012/116602

27 *Methods with regard to the risk of dropped loads for UK EPR for concrete structure.* ENGSGC100483 Revision B, EDF, February 2012. TRIM Ref. 2012/116384.

28 *Step 3 Internal Hazards Assessment of the EDF and AREVA UK EPR*. HSE-ND Assessment Report AR 09/026. November 2009. TRIM Ref. 2009/330030.

29 *The Lifting Operations and Lifting Equipment Regulations 1998*. Statutory Instrument 1998 No. 2307, December 1998.

30 *The Management of Health and Safety at Work Regulations 1992.* Statutory Instrument 1992 No. 2051, January 2003.

31 *Design of Lifting Equipment in Nuclear Power Plants.* KTA3902, Nuclear Safety Standards Commission, June 1999.

32 *High Safety Lifting and Handling Machines*, BTS 60.C.007.03, EDF.

33 *EDF and AREVA UK EPR - Schedule of Regulatory Observations Raised during Step 4.* HSE-ND. TRIM Ref. 2010/600727.

34 *"UK EPR GDA – Management of Nuclear Safety Significant Lifting.* ECEMA101802, Revision B, EDF, December 2010. TRIM Ref. 2011/85921

35 *Single-Failure Proof Cranes for Nuclear Power Plants*. NUREG-0554, U.S. Nuclear Regulatory Commission, May 1979.

36 *Control of Heavy Loads at Nuclear Power Plants Resolution of Generic Technical Activity A-36*. NUREG-0612, U.S. Nuclear Regulatory Commission, July 1980.

37 *Step 4 Mechanical Engineering Assessment of the EDF and AREVA UK EPR™ Reactor*. ONR Assessment Report ONR-GDA-AR-11-026 Revision 0. TRIM Ref. 2010/581505. (in TRIM folder 4.4.1.1827.).

38 *Step 4 Civil Engineering and External Hazards Assessment of the EDF and AREVA UK EPR™ Reactor*. ONR Assessment Report ONR-GDA-AR-11-018 Revision 0. TRIM Ref. 2010/581513. (in TRIM folder 4.4.1.1827.).

39 *GDA Close-out for the EDF and AREVA UK EPR™ Reactor – GDA Issue GI-UKEPR-CE-02 Revision 1 – Use of ETC-C for the Design and Construction of the UK EPR™* ONR Assessment Report ONR-GDA-AR-12-004, Revision 0, TRIM 2012/4.

40 PCSR Sub-Chapter 13.2 Update – Internal Hazards Protection, UKEPR-0002-132 Issue 04, 13th September 2012, TRIM 2012/367280.

Table 1

Relevant Safety Assessment Principles Considered for Close-out of GI-UKEPR-IH-01 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-01 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.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.
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-01 Revision 2 – Substantiation and analysis of the consequences of dropped loads and impact from lifting equipment included within the UK EPR™ design – EDF and AREVA Deliverables

GDA Issue Action	Internal Hazards Topic	Document Ref.	Title	Ref.
GI-UKEPR-IH-01.A1	Dropped Loads and Impact	ECEIG111683 Revision A	*Dropped Loads – Summary of Design Basis and Principles.*	16
GI-UKEPR-IH-01.A1	Dropped Loads and Impact	PEPS-G/2011/en/1060 Revision A	*Identification of Representative Drop Load Cases from the Safety Class 1 Polar Crane in the Reactor Building.*	17
GI-UKEPR-IH-01.A1	Dropped Loads and Impact	ECEIG111791 Revision A	*Identification of Representative Drop Load Cases from the Safety Class 2 Cranes*	18
GI-UKEPR-IH-01.A1	Dropped Loads and Impact	ECEIG120274 Revision A	*ECEIG120274 A, Summary Report for the Substantiation of "Dropped Loads" Hazard, February 2012*	19
GI-UKEPR-IH-01.A1	Dropped Loads and Impact	ECEIG120198 Revision A	*ECEIG120198 A, Safety Case for 4 Representative Load Drops from Safety Classified 2 Cranes, February 2012*	20
GI-UKEPR-IH-01.A1	Dropped Loads and Impact	ECEIG111395 Revision A	*ECEIG111395 A, Application Note for a Drop Load Impact on a Reinforced Concrete Slab, March 2012*	21
GI-UKEPR-IH-01.A1	Dropped Loads and Impact	ECEIG111620 Revision A	*ECEIG111620 A, EPR UK – RS2 cranes – Drop Load Impact Calculations, January 2012*	22
GI-UKEPR-IH-01.A1	Dropped Loads and Impact	PEPS-G/2011/en/1076 Revision C	*PEPS-G/2011/en/1076 C – ALARP justification of Identified Representative Drop Load Cases from the Safety Class 1 Polar Crane in the Reactor Building, February 2012*	23
GI-UKEPR-IH-01.A1	Dropped Loads and Impact	PEER-F DC 71 Revision B	*PEER-F DC 71 B, Consequences on the Reactor of an Accidental RPV Head Drop During it's Handling, February 2012*	24

Annex 1

Deliverables and Associated Technical Queries Raised During Close-out Phase

GI-UKEPR-IH-01 Revision 2 – Substantiation and analysis of the consequences of dropped loads and impact from lifting equipment included within the UK EPR™ design – EDF and AREVA Deliverables

GDA Issue Action	Internal Hazards Topic	Document Ref.	Title	Ref.
GI-UKEPR-IH-01.A1	Dropped Loads and Impact	PEPR–F DC 85 Revision B	*PEPR–F DC 85 B, Drop of a Reactor Cavity Cover Slab on the RPV Closure Head Analysis, February 2012*	25
GI-UKEPR-IH-01.A1	Dropped Loads and Impact	PECS-G/2011/en/0018 Revision B	*PECS-G/2011/en/0018 B, Check of Bearing Capacity of Reinforced Concrete Reactor Pool Slab Subject to Drop Load of a Concrete Cover Slab and a Multi-Stud Tensioning Machine, February 2012*	26
GI-UKEPR-IH-01.A2	Dropped Loads and Impact	ENGSGC100483 Revision A	*Methods with regard to the risk of dropped loads for UK EPR for concrete structure*	27

GI-UKEPR-IH-01 Revision 2 – Substantiation and analysis of the consequences of dropped loads and impact from lifting equipment included within the UK EPR™ design – Technical Queries Raised

TQ Reference	GDA Issue Action	Related Submission	Description
None.			

Annex 2

GDA Assessment Findings Arising from GDA Close-out for Internal Hazards GDA Issue GI-UKEPR-IH-01

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-IH-9	The Licensee shall ensure that the further studies in order to support the design modification associated with the manual connection of the LHSI/RHR system are appropriately considered within the site specific design.	*Mechanical, Electrical, and C&I 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-01 – Internal Hazards – UK EPR™

EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT

GDA ISSUE

SUBSTANTIATION AND ANALYSIS OF THE CONSEQUENCES OF DROPPED LOADS AND IMPACT FROM LIFTING EQUIPMENT INCLUDED WITHIN THE EPR DESIGN

GI-UKEPR-IH-01 REVISION 2

<table>
<tr><td colspan="2">Technical Area</td><td colspan="2">INTERNAL HAZARDS</td></tr>
<tr><td colspan="2">Related Technical Areas</td><td colspan="2">Mechanical Engineering
Civil Engineering</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-UKEPR-IH-01</td><td>GDA Issue Action Reference</td><td>GI-UKEPR-IH-01.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">Substantiation and analysis of the consequences of dropped loads and impact from lifting equipment included within the EPR design.</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide substantiation of the nuclear safety significant structures, systems and components vulnerable to dropped load and impact from RS1 and RS2 lifting equipment.

It is the expectation of ONR that dropped loads be considered for lifts that may result in nuclear significant consequences. The response should include detailed assessment of potential loads that could be dropped under such conditions and demonstrate that the provisions in place to ensure that the risk to nuclear safety of a load drop or impact is ALARP. Such assessment may include multi-legged arguments which consider the following:
Claims on civil structures.
Additional physical protection.
Limits and conditions on the use of the RS1 and RS2 lifting equipment.
Provision of detailed load path routes avoiding areas of highest nuclear significance.
Measures (both system based and administratively controlled) in place to ensure the potential for impact of the load is minimised.
Any further defence in depth and ALARP measures that could be implemented into the design.
The impact of the changes made to the PCSR relating to the outcome of this substantiation on other safety case submissions submissions.
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-01 – Internal Hazards – UK EPR™

EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT

GDA ISSUE

SUBSTANTIATION AND ANALYSIS OF THE CONSEQUENCES OF DROPPED LOADS AND IMPACT FROM LIFTING EQUIPMENT INCLUDED WITHIN THE EPR DESIGN

GI-UKEPR-IH-01 REVISION 2

<table>
<tr><td colspan="2">Technical Area</td><td colspan="2">INTERNAL HAZARDS</td></tr>
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Civil Engineering</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-UKEPR-IH-01</td><td>GDA Issue Action Reference</td><td>GI-UKEPR-IH-01.A2</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide a description of the approach taken to treat dropped loads on civil structures, including consideration of the following:
• Derivation of design loads.
• Analysis methods.
• Design rules.
• Reliability expectations.
• Consistency between ECEIG070272 REV A1 “EPR- Load Drops - Methodology for risk analysis in civil engineering and building installations - Design review preparation conditions” and ETC-C in relation to consideration of Global stability.
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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