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-CE-05 Revision 1 – Reliability of the ETC-C

Assessment Report: ONR-GDA-AR-12-001
Revision 1
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EXECUTIVE SUMMARY

This report presents the close-out of the Health and Safety Executive's (HSE) Generic Design Assessment (GDA) within the area of Civil Engineering and External Hazards. The report specifically addresses the GDA Issue **GI-UKEPR-CE-005 Revision 1** and associated GDA Issue Actions generated as a result of the GDA Step 4 Civil Engineering and External 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.

Each of the Civil Structures in the reference design has been designed using the EPR Technical Code-Civil (ETC-C). This is an EDF and AREVA specific code, developed for the EPR project. The ETC-C is intended to adapt Eurocodes and other relevant standards and is essentially a signposting document which directs the designer to assorted Eurocodes, European Standards (EN), French standards and other guidance.

One aspect which required further justification which was raised during Step 4 of the Health and Safety Executive's (HSE) Generic Design Assessment (GDA) was the reliability of the ETC-C as a design code; in other words how confident can we be that structures designed to it will meet the safety demands placed upon them. The background to the Eurocodes also states that "*For the design of special construction works (e.g. nuclear installations, dams, etc.) other provisions than those in the EN Eurocodes might be necessary*". This statement reflects the higher demands placed on nuclear structures, and that they should have a higher safety consideration than standard industrial or commercial buildings.

The responses provided in GDA identified the two most critical areas as the design of the containment against seismic loading and against overpressure. The detailed calculations of the achieved reliabilities submitted during GDA were not found to be fully satisfactory and a GDA Issue was raised (**GI-UKEPR-CE-05 Revision 1**).

The responses provided to **GI-UKEPR-CE-05 Revision 1** have been found to be satisfactory, and demonstrate that the ETC-C does ensure that the necessary reliability can be achieved through its appropriate use. Two assessment findings have been raised for implementation by a future licensee.

LIST OF ABBREVIATIONS

BMS	(ONR) How2 Business Management System
CMF	Change Modification Form
EDF and AREVA	Electricité de France SA and AREVA NP SAS
EPRI	Electric Power Research Institute
ETC-C	EPR Technical Code - Civil
FORM	First Order Reliability Method
GDA	Generic Design Assessment
HSE	The Health and Safety Executive
ONR	Office for Nuclear Regulation
PCSR	Pre-construction Safety Report
RO	Regulatory Observation
SAP	Safety Assessment Principles (ONR)
SORM	Second Order Reliability Method
TAG	Technical Assessment Guide (ONR)

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1 INTRODUCTION

1.1 Background

1 This report presents the close-out of the Health and Safety Executive's (HSE) Generic Design Assessment (GDA) within the area of Civil Engineering and External Hazards. The report specifically addresses the GDA Issue **GI-UKEPR-CE-005 Revision 1** and associated GDA Issue Actions (Ref. 32) generated as a result of the GDA Step 4 Civil Engineering and External Hazards Assessment of the UK EPR™ (Ref. 4). The assessment has focussed on the deliverables identified within the EDF and AREVA Resolution Plan (Ref. 33) 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 the 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 Civil Engineering and External Hazards Assessment identified a number of GDA Issues and Assessment Findings as part of the assessment of the evidence associated with the UK EPR™ reactor design. A GDA Issue is an observation of particular significance that requires resolution before the Office for Nuclear Regulation (ONR), an agency of HSE, would agree to the commencement of nuclear safety related construction of the UK EPR™ within the UK. An Assessment Finding results from a lack of detailed information which has limited the extent of assessment and as a result the information is required to underpin the assessment. However, they are to be carried forward as part of normal regulatory business.

4 The Step 4 Assessment concluded that the UK EPR™ reactor was suitable for construction in the UK subject to resolution of 31 GDA Issues. The purpose of this report is to provide the assessment which underpins the judgement made in closing GDA Issue **GI-UKEPR-CE-005**.

5 Each of the Civil Structures in the reference design has been designed using the EPR Technical Code-Civil (ETC-C), Ref. 5. This is an EDF and AREVA specific code, developed for the EPR project. The ETC-C is intended to adapt Eurocodes and other relevant standards and is essentially a signposting document which directs the designer to assorted Eurocodes, European Standards (EN), French standards and other guidance.

6 One area which was raised as a Regulatory Observation (RO) in Step 4 of GDA (**RO-UKEPR-037**) was the reliability of the ETC-C as a design code, in other words how confident can we be that structures designed to it will meet the safety demands placed upon them.

7 The background to the Eurocodes also states that "*For the design of special construction works (e.g. nuclear installations, dams, etc.) other provisions than those in the EN Eurocodes might be necessary*". This statement reflects that some of the Eurocode rules should be amended and/or extended to reflect the specific demands placed on nuclear structures, and that they should have a higher safety consideration than standard industrial or commercial buildings. The other fundamental tenet of the Eurocodes is that there is the option to select the levels of reliability required through appropriate choice of not only design methods (partial factors), but also implementation control methods.

8 The response to **RO-UKEPR-037** Action 1, which requested clarity over the required reliability assured by the use of the ETC-C, identified the two most critical areas as the

design of the containment against seismic loading and against overpressure along with the target reliabilities.

9 Details over the required and achieved reliabilities for the containment were provided as part of the submissions against **RO-UKEPR-37**.

10 These initial submissions to **RO-UKEPR-037** were found to fall short of our expectations, and ONR wrote to EDF and AREVA (Ref. 6), with detailed comments. The response to these comments did not arrive in sufficient time to be included in the Step 4 assessment, and as a result, a GDA Issue was raised.

1.2 Scope

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

12 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 highlighted and addressed as part of the close-out phase or be identified as Assessment Findings to be taken forward to site licensing.

13 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.

1.3 Methodology

14 The methodology applied to this assessment is identical to the approach taken during Step 4 which followed the ONR BMS document AST/001, Assessment Process (Ref. 1), in relation to mechanics of assessment within ONR.

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

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

1.4 Structure

17 This Assessment Report structure differs slightly from the structure adopted for the previous reports produced within GDA, most notably the Step 4 Civil Engineering and External Hazards Assessment (Ref. 4). The report has been structured with the assessment of the individual GDA Issue rather than a report detailing close out of all GDA Issues.

18 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 GDA ISSUE GI-UKEPR-CE-05

19 The intended assessment strategy for GDA close-out for the Civil Engineering and External Hazards topic area was set out in an assessment plan that identified the intended scope of the assessment and the standards and criteria that would be applied.

20 The overall basis for the assessment of the GDA Issues are the Civil Engineering and External hazards elements of:

- Submissions made to ONR in accordance with the Resolution Plans.
- Update to the Submission / Pre-construction Safety Report (PCSR) / Supporting Documentation.
- The Design Reference that relates to the Submission / PCSR as set out in UK EPR™ GDA Project Instruction **UKEPR-I-002 Revision 11** (Ref. 31) which will be updated throughout GDA Issue resolution. This includes Change Management Forms (CMF), and any A2 and B Category design changes agreed for inclusion into GDA.
- Design Change Submissions – which are proposed by EDF and AREVA and submitted in accordance with UK-EPR GDA Project Instruction UKEPR-I-003 (Ref. 34).

2.1 The Approach to Assessment for GDA Close-out

21 The approach to the closure of the GDA for the UK EPR™ Project involves the assessment of submissions made by EDF and AREVA in response to GDA Issues identified through the GDA process. These submissions are detailed within the EDF and AREVA Resolution Plan to the GDA Issue.

2.2 Standards and Criteria

22 The relevant standards and criteria adopted within this Assessment are principally the Safety Assessment Principles (SAP), internal technical assessment guides, relevant national and international standards and relevant good practice informed from existing practices adopted on UK nuclear licensed sites. The key SAPs and relevant ONR Technical Assessment Guides (TAG) 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 assessment of GDA Issue **GI-UKEPR-CE-05** are identified below, reproduced from Ref. 2.

"

Engineering principles: safety classification and standards	***Standards***	***ECS.3***
Structures, systems and components that are important to safety should be designed, manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected to the appropriate standards.		

157 The standards should reflect the functional reliability requirements of structures, systems and components and be commensurate with their safety classification.

158 Appropriate national or international codes and standards should be adopted for Classes 1 and 2 of structures, systems and components. For Class 3, appropriate non-nuclear-specific codes and standards may be applied.

159 Codes and standards should be preferably nuclear-specific codes or standards leading to a conservative design commensurate with the importance of the safety function(s) being performed. The codes and standards should be evaluated to determine their applicability, adequacy and sufficiency and should be supplemented or modified as necessary to a level commensurate with the importance of the safety function(s) being performed.

160 Where a structure, system or component is required to deliver multiple safety functions, and these can be demonstrated to be delivered independently of one another, codes and standards should be used appropriate to the category of the safety function. Where independence cannot be demonstrated, codes and standards should be appropriate to the class of the structure, system or component (i.e. in accordance with the highest category of safety function to be delivered). Whenever different codes and standards are used for different aspects of the same structure, system or component, the compatibility between these should be demonstrated.

161 The combining of different codes and standards for a single aspect of a structure, system or component should be avoided or justified when used. Compatibility between these codes and standards should be demonstrated.

Engineering principles: reliability claims	***Form of claims***	***ERL.1***
The reliability claimed for any structure, system or component important to safety should take into account its novelty, the experience relevant to its proposed environment, and the uncertainties in operating and fault conditions, physical data and design methods.		

“

2.2.2 Technical Assessment Guides

24 The following technical assessment guides have been used as part of this assessment:

- ONR BMS. Structural Integrity Civil Engineering Aspects. T/AST/017 Issue 2 (Ref. 3).
- ONR BMS. External Hazards. T/AST/013 Issue 3 (Ref. 3).

2.2.3 National and International Standards and Guidance

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

- EPRI Guidance Documents (Refs 22 to 29).

2.3 Use of Technical Support Contractors

26 Assistance to ONR has been provided by Atkins. They have in turn employed Safety and Reliability Consultants Ltd as specialist support.

2.4 Out-of-scope Items

27 There are no out of scope items. The entirety of GDA Issue **GI-UKEPR-CE-05 Revision 1** has been addressed. In addition, there are no changes to the scope of the GDA assessment detailed in the Step 4 report (Ref. 4).

3 GDA ISSUE

3.1 Background to the GDA Issue and Associated GDA Issue Actions

28 Regulatory Observation 37 (**RO-UKEPR-037**) was raised during Step 4 of GDA on the reliability of the ETC-C as a design code, in other words how confident can we be that structures designed to it will meet the safety demands placed upon them. The background to the Eurocodes also states that *"For the design of special construction works (e.g. nuclear installations, dams, etc) other provisions than those in the EN Eurocodes might be necessary"*. This statement reflects the higher demands placed on nuclear structures, and that they should have a higher safety consideration than standard industrial or commercial buildings. The other fundamental tenet of the Eurocodes is that there is the option to select the levels of reliability required through appropriate choice of not only design methods (partial factors), but also implementation control methods.

29 Responses to **RO-UKEPR-037** were received during Step 4, but were not considered to be satisfactory and ONR wrote to EDF and AREVA (Ref. 6), with detailed comments. These were not addressed in sufficient time to allow inclusion within the Step 4 assessment. As a result, GDA Issue **GI-UKEPR-CE-05 Revision 1** was raised at the end of the Step 4 process.

3.2 EDF and AREVA Deliverables in Response to the GDA Issue

30 The information provided by EDF and AREVA in response to this GDA Issue is contained in Refs 8 to 19.

31 It is important to note that some of this information is supplementary to the information provided within the PCSR (Ref. 30) which has already been subject to assessment during earlier stages of GDA. In addition, it is important to note that the deliverables are not intended to provide the complete safety case for Civil Engineering and External Hazards Area. Rather, they form further detailed arguments and evidence to supplement those already provided during earlier Steps within the GDA Process.

3.3 ONR Assessment

32 Further to the assessment work undertaken during Step 4 (Ref. 4) and the resulting GDA Issue, **GI-UKEPR-CE-05**, this assessment focuses on reviewing Refs 8 to 19.

33 This assessment has been carried out in accordance with the ONR BMS document AST/001, "*Assessment Process*" (Ref. 1).

3.4 Scope of Assessment Undertaken

34 The scope of the assessment has been to consider the expectations detailed down with the GDA Issue, **GI-UKEPR-CE-05 Revision 1**, and the associated GDA Issue Actions. These are detailed within Annex 2 of this report. For each of the following areas further evidence was sought:

- Demonstration that the ETC-C provides the required level of reliability for the most safety critical structures and load-cases.

35 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 civil engineering and external hazards such that the existing case is undermined, these have been addressed.

3.5 Assessment

3.5.1 General Comments on the Submissions

36 The revised submissions (Refs 8 to 19) provide a much greater level of detail and justification for the approach used. In particular, the overpressure case is much more clearly presented than previously.

3.5.2 Reliability under Seismic Loading

37 The approach chosen by EDF and AREVA to assess the structural reliability has been to use the methodology developed over about two decades by the US Electric Power Research Institute (EPRI) (Refs 22 to 29). This approach is broadly acceptable; however there are some points worthy of mention.

38 The EPRI method is based on a number of assumptions – the most important of which is that the structural capacity is lognormally distributed. There will, however, be situations where this is not true, depending on the nature of the mathematical expression defining the structural capacity. The examples used in the current studies appear to give a reasonable range of failure modes, however there may be examples which are not considered here which have a different distribution.

39 The initial response to **RO-UKEPR-037 Action 1** derived the reliability requirements for the inner containment against seismic and overpressure loading. From the work presented in Reference 9 by EDF and AREVA it can be concluded that the UK EPR™ meets the seismic reliability target of about 1×10^{-8}. To test this claim, the probability of failure was obtained by carrying out an independent numerical integration of the convolution integral gave results very similar to the value obtained by EDF and AREVA, thus confirming the numerical accuracy of their calculations (Ref. 21).

40 It should be noted that one of the reasons for the apparent robustness of the EPRI methodology for seismic reliability analysis, in spite of its various approximations, is that the hazard function is highly skewed. This results in the computed failure probability being more sensitive to the seismic loading than to the structural capacity. For the calculations undertaken it has been shown that the peak contribution to failure probability occurs at about 1.2g which corresponds to about the 1% fractile of the fragility curve (i.e. not a very extreme value, and one which can therefore be calculated with reasonable accuracy). This can be contrasted with the situation of thermal-hydraulic loading where failure can only occur at extremely low values of structural capacity if the maximum internal pressure in the containment is assumed to be capped at the design pressure of 0.65 MPa gauge.

41 The demonstration in Reference 9 is based on the design for Flamanville, which is a very hard site. Site specific conditions will be different, (both soil conditions and seismic demand) and almost certainly softer, and as a result, the interaction between the raft and the inner containment will be different, especially as the raft will undergo design changes. Thus, although it is clear that the ETC-C has delivered adequate levels of reliability for the generic design, the use of these reliabilities for safety claims for the UKEPR will require further justification.

42 For future reliability calculations involving the UK EPR™ under seismic loading, it is recommended that more advanced methods should be used if reliance is placed on demonstration of reliabilities with certain levels of accuracy. Methods such as directional simulation should be used in conjunction with more advanced structural response analysis. This is an assessment finding. The Licensee shall demonstrate to an acceptable level of confidence any claims made on the reliability of the containment under seismic loading. These claims shall be supported using modern methods of simulation such as FORM and SORM. This shall take into account the design process undertaken, and the variation in strengths achieved in the construction of the containment.. This shall be undertaken ahead of the first pressure test (Assessment Finding **AF-UKEPR-CE-69**).

> ***AF-UKEPR-CE-69:*** *The Licensee shall demonstrate to an acceptable level of confidence any claims made on the reliability of the containment under seismic loading. These claims shall be supported using modern methods of simulation such as FORM and SORM. This shall take into account the design process undertaken, and the variation in strengths achieved in the construction of the containment.*
>
> **Required timescale:** Ahead of the first containment pressure test

3.5.3 Reliability under Overpressure

43 The approach chosen by EDF and AREVA to assess the structural reliability has been to use the methodology developed over about two decades by the US Electric Power Research Institute (EPRI) (Refs 22 to 29). This approach is primarily intended for seismic reliability. The use of a truncated hazard range for within design basis events (no greater demand than 0.65 MPa gauge) and the associated simplification of the calculations used required further justification. Simply put, it was not considered that using a methodology with such a large number of assumptions (conservative and un-conservative) to derive very low probabilities of failure was fully justified.

44 In response to the ONR letter (Ref. 6), a series of revised and new documents have been produced (Refs 8 to 19). The contents of the key documents are examined in more detail below.

AREVA and EDF. ENGSGC100106 rev B – Study of the behaviour of the EPR inner containment wall beyond design-basis conditions. 18/02/2011 (Ref. 13)

45 This revised document is much clearer and addresses the bulk of the comments raised. In particular, it clarifies that the probability of failure being calculated is the conditional probability of failure given that the pressure in the inner containment structure does not exceed 0.65 MPa gauge.

46 The statements in ENGSGC100106 that *"The probability of failure [of the inner containment] is estimated at $6x10^{-11}$ per reactor year"* are not entirely accurate. A more correct statement would be that *"The probability of failure of the inner containment is estimated at $6x10^{-11}$ per reactor* for events considered within the design basis in which the internal pressure does not exceed the design pressure of 0.65MPa gauge*"*. It is suggested that the PCSR should be updated to include this clarification.

47 The justification of all of the uncertainties used is still not seen as fully robust. Those used to capture analysis and modelling variability are seen as adequate, however those for material properties are seen as somewhat optimistic. This is also linked to the requirements placed on the construction process in Part 2 of ETC-C AFCEN (Ref. 5). Evidence is presented to support the values of uncertainty of pre-stressing strand based upon French data, and the values appear to be consistent with the evidence collected. It is not fully clear that the ETC-C will guarantee that such low levels of variability will be achieved for a UK EPR™. There is confidence that the characteristic will be met, however the multiplier to mean and the resulting standard deviation may well be different, especially if more than one supplier is used. Similar comments apply to the concrete and reinforcement.

48 The importance of the material variability to the overall variability varies depending on the location within the containment. For the most critical areas such as the personnel hatch and equipment hatch, the variability is governed strongly by the modelling variability, and the materials variability has a second order effect. For other less critical areas such as the main belt area, the overall variability is equivalently dependent on the materials, and modelling variability. In these areas however the margin is significantly higher and the probabilities of failure consequently lower. There is therefore a high degree of confidence that changes in the material variability at a site level will not affect the overall failure probabilities sufficient to reduce them below the target values provided the rules given in ETC-C part 2 are followed.

49 The overall predicted frequency of failure under overpressure is estimated to be 6×10^{-11}. The risk target from of RO-UKEPR-037 Action 1 is 6×10^{-9}. The reliability obtained by the generic design is therefore demonstrated to be acceptable. It is also clear that there is margin between the target and the prediction to accommodate potential changes in the levels of reliability achieved for the material properties and still achieve the overall reliability required.

50 It is considered that it would be possible for EDF and AREVA to undertake parametric studies which demonstrate that the overall reliability achieved is relatively insensitive to the assumptions made over the material coefficients of variability and that there is clear evidence that the ETC-C Part 2 will ensure that the materials used fall within these boundaries. For the reasons given above however there is a high confidence that this will be the case, provided the materials are within the ETC-C specifications. Therefore, a more practicable approach is to develop an assessment finding which requires the Licensee to demonstrate that the achieved variabilities in strengths are sufficient to provide appropriate levels of reliability for the UK EPR™. During the construction phase of the containment, there will be extensive material testing which will provide a high degree of confidence in both the mean properties obtained and the variability around the mean. Confirmation of the achieved reliabilities cannot be made until the construction of the containment is complete, as the data from the prestressing strand will only be available late in the construction timeframe.

51 The Licensee shall confirm through appropriate simulation that the reliability of the containment structure against overpressure satisfies the safety case requirements. This shall take into account the design process undertaken, and the variation in strengths achieved in the construction of the containment. In addition, a full range of failure scenarios shall be considered. This shall be undertaken ahead of the first pressure test (Assessment Finding **AF-UKEPR-CE-70**).

***AF-UKEPR-CE-70:** The Licensee shall confirm through appropriate simulation that the reliability of the containment structure against overpressure satisfies the safety case requirements. This shall take into account the design process undertaken, and the variation in strengths achieved in the construction of the containment. In addition, a full range of failure scenarios shall be considered.*

Required timescale: Ahead of the first containment pressure test

Coyne et Bellier. 12 680 RP 01-41 rev A – Answer to HSE Regulatory Observation RO-UKEPR-037 – Ultimate Pressures in EPR Containment – Comparison of simplified method (EPRI method) to method based on statistical numerical simulation. 06/12/2010 (Ref. 18)

52 The report aims to investigate an issue dealing with the statistical distribution of the containment ultimate pressure fragility curves, as raised in **RO-UKEPR-037** by ONR. In order to try to answer the question as to whether the fragility curves can be assumed to follow lognormal distributions Coyne et Bellier have made comparisons *"for a selected number of the main containment failure modes, between the results obtained…"* by *"…using the simplified EPRI method with an a priori assumption of lognormally distributed failure pressure (calculation method **C1**)"* and the results obtained from a *"…full statistical numerical simulation using as input data N sets of random lognormally distributed component variables (calculation method **C2**)"*.

53 Analysis has shown that the conclusion reached that the probability distribution of the ultimate pressure capacity of the inner containment can be modelled by lognormal distributions in each of the three different failure modes considered is sufficiently correct for the 95% confidence value P_{ult},0.95 to be calculated with minimal error using the EPRI approach.

54 However, the reason for P_{ult} being very close to lognormal for all three failure modes is that the overall variability in P_{ult} is dominated in each case by a single lognormal variable, such that all the other uncertainty contributions are relatively unimportant. If this were not the case and the uncertainties were more evenly distributed between the variables the distribution of P_{ult} would lie somewhere between a normal and a lognormal distribution, and a lognormal assumption would be potentially un-conservative.

55 Furthermore, the coefficients of variation of all the input variables have been taken to be so low that the resulting coefficient of variation (or logarithmic standard deviation) of P_{ult} is also very low (typically about 2%) with the result that the lognormal model differs only a small amount from a normal distribution, except in the extreme tails.

56 It cannot therefore be concluded from the studies undertaken that the EPRI approach will produce conservative results under all circumstances, even though this method is acceptable for the particular failure modes selected. For the case of the inner containment, I consider the failure modes examined encompass the credible modes of failure for a structure of this type, and the approach is therefore seen to be sufficient to demonstrate at a generic level that adequate reliabilities can be achieved.

Coyne et Bellier. 12 680 RP 01-45 rev B – Answer to HSE Regulatory Observation RO-UKEPR-037 – Determination of failure mode ultimate pressures – comparison of simplified (EPRI) with fully statistical methods – Special case of low probabilities. 04/01/11 (Ref. 19)

57 The aim of this report by Coyne et Bellier was to investigate further the differences in the computed failure probabilities obtained when using the approximate EPRI method and

those obtained from a more exact method based on Monte Carlo simulation, especially for low probability failure events.

58 From the information given in the review of 12 680 RP 01-41, the results of this report can easily be predicted in advance. The aim was to investigate the differences between the results of the approximate EPRI method, in which it is explicitly assumed that the distribution of P_{ult} (the capacity of the containment to resist internal pressure) is lognormal, and a more accurate simulation method. However, as demonstrated above both the failure modes considered are dominated by a single random variable (the force in the pre-stressing cables in the case of cylinder hoop failure; and the height of the failure plane above the level of the gusset base, h_{strut}, in the case of shear failure). This dominance and the fact that it is assumed *a priori* that both these input variables are lognormally distributed means that the distribution of P_{ult} is likely to be close to lognormal. This means that the differences between the two approaches are likely to be small for the two failure modes considered.

59 One criticism of the calculations performed is that in their “statistical numerical simulation” they have used only 1×10^{6} trials and yet are trying to estimate P_{ult} at probabilities of non-exceedance as low as 1×10^{-9}. This has been attempted by fitting a lognormal distribution to the simulation output (their so-called adjusted distribution). However, as part of the aim of their study was to check whether P_{ult} could be treated as lognormal, the fitting of a lognormal distribution to the simulation output severely clouds the issue. To achieve sufficiently high accuracy in a basic simulation approach at a probability level of 1×10^{-9} would require approximately 1×10^{11} trials, which is not likely to be feasible.

60 An independent calculation using First Order Reliability Method (FORM) and Second Order Reliability Method (SORM) has been made (Ref. 21) for the gusset shear failure mode. Using this approach, the capacity to resist a pressure P_{ult} corresponding to a probability of non-exceedance of 1×10^{-9} has been calculated as $P_{ult}(1 \times 10^{-9}) =$ 1.016 MPa. By way of comparison, the simplified Monte Carlo approach used in this report gives 0.908 MPa and the EPRI method gives 0.580 MPa. These are very significant differences. However, in this case, the EPRI method is conservative as it predicts failure occurring at significantly lower pressures than do the more rigorous calculation methods. The general conclusion that must be drawn is that the EPRI method is not a fully rigorous approach, but that in the cases examined it does give conservative results. The failure modes examined represent a broad cross section of failure modes that are considered to encompass the most critical cases, and are therefore sufficient to demonstrate at a generic level that the adequate reliabilities can be achieved. It is my judgement supported by the ancillary work undertaken that the approach adopted is sufficiently conservative. This is further supported by the margin demonstrated between the required and predicted reliabilities.

4 ASSESSMENT CONCLUSIONS

61 The same conclusions are drawn from a review of all the revised submissions (Refs 8 to 19). Within the bounds of the assumptions made they give a reasonable demonstration of the low probability of failure of the inner containment against seismic loading and against the postulated limiting internal pressure of 0.65 MPa Gauge. In this regard, the requirements of SAPs ECS.3 and ERL.1 are satisfied. This is considered sufficient for the purposes of the Generic Design Assessment, however there are a number of aspects of the approach where there are remnant uncertainties as a result of the simplifications adopted. Independent calculations have shown that for some cases there is good agreement but for others this is less evident. It is considered likely that the approach adopted will always err on the conservative side for most practical cases, however this has not been proven unequivocally.

62 It has therefore been decided to generate two assessment findings (**AF-UKEPR-CE-69** and **AF-UKEPR-CE-70)** which will provide confidence in the reliabilities obtained through the design and construction process.

5 ASSESSMENT FINDINGS

63 I conclude that the following Assessment Findings, also listed in Annex 1, should be programmed during the forward programme of this reactor as normal regulatory business, in addition to those identified in the Step 4 Civil Engineering Assessment Report, Ref. 4.

5.1 Additional Assessment Findings

> ***AF-UKEPR-CE-69:*** *The Licensee shall demonstrate to an acceptable level of confidence any claims made on the reliability of the containment under seismic loading. These claims shall be supported using modern methods of simulation such as FORM and SORM. This shall take into account the design process undertaken, and the variation in strengths achieved in the construction of the containment.*

> ***AF-UKEPR-CE-70:*** *The Licensee shall confirm through appropriate simulation that the reliability of the containment structure against overpressure satisfies the safety case requirements. This shall take into account the design process undertaken, and the variation in strengths achieved in the construction of the containment. In addition, a full range of failure scenarios shall be considered*.

5.2 Impacted Step 4 Assessment Findings

64 There are no impacted Step 4 Assessment Findings.

6 REFERENCES

1 *ONR How2. Assessment Process.* AST/001 Issue 4. HSE. April 2010. www.hse.gov.uk/nuclear/operational/assessment/index.htm.

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

3 *Structural Integrity Civil Engineering Aspects*. T/AST/017 Issue 2. HSE. March 2005
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Annex 1

GDA Assessment Findings Arising from GDA Close-out for Civil Engineering and External Hazards GDA Issue GI-UKEPR-CE-05 Revision 1

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-CE-69	*The Licensee shall demonstrate to an acceptable level of confidence any claims made on the reliability of the containment under seismic loading. These claims shall be supported using modern methods of simulation such as FORM and SORM. This shall take into account the design process undertaken, and the variation in strengths achieved in the construction of the containment.*	Ahead of the first containment pressure test.
AF-UKEPR-CE-70	The Licensee shall confirm through appropriate simulation that the reliability of the containment structure against overpressure satisfies the safety case requirements. This shall take into account the design process undertaken, and the variation in strengths achieved in the construction of the containment. In addition, a full range of failure scenarios shall be considered.	Ahead of the first containment pressure test.

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 during the operational phase. For other Assessment Findings, it is the regulators' expectation that the findings are adequately addressed no later than the milestones indicated above.

Annex 2

GDA Issue, GI-UKEPR-CE-05 Revision 1

EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT

GDA ISSUE

RELIABILITY OF THE ETC-C

GI-UKEPR-CE-05 REVISION 1

<table>
<tr><td colspan="2">Technical Area</td><td colspan="2">CIVIL ENGINEERING</td></tr>
<tr><td colspan="2">Related Technical Areas</td><td colspan="2">None</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-UKEPR-CE-05</td><td>GDA Issue Action Reference</td><td>GI-UKEPR-CE-05.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">There is not yet sufficient demonstation of the reliabilities achieved by use of the ETC-C as a design code.</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Support assessment within the following areas and provide adequate responses to any questions arising from the assessment by ONR of submissions received late in Step 4 of GDA around the following topics:
• Reliability of EPR Inner Containment to earthquake.
• Target reliabilities for UK EPR structures built to ETC-C.
• Behaviour of EPR Inner Containment wall beyond design-basis conditions.
Based on a high level review of the documents and assurances provided to date I have sufficient confidence in the approach to conclude that it should be possible to provide a suitable demonstration of both the beyond design basis performance and the fragility for use in the PSA.
With agreement from the Regulator this action may be completed by alternative means.</td></tr>
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