Office for Nuclear Regulation
An agency of HSE

Generic Design Assessment – New Civil Reactor Build

Step 4 Civil Engineering and External Hazards Assessment of the EDF and AREVA UK EPR™ Reactor

Assessment Report: ONR-GDA-AR-11-018
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PREFACE

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

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

EXECUTIVE SUMMARY

This report presents the findings of the Civil Engineering and External Hazards assessment of the UK EPR reactor undertaken as part of Step 4 of the Health and Safety Executive's (HSE) Generic Design Assessment (GDA). The assessment has been carried out on the November 2009 Pre-construction Safety Report (PCSR) and supporting documentation submitted by EDF and AREVA during Step 4.

This assessment has followed a step-wise-approach in a claims-argument-evidence hierarchy. In Step 2 the claims made by EDF and AREVA were examined, in Step 3 the arguments that underpin those claims were examined.

The scope of the Step 4 assessment was to review the safety aspects of the UK EPR reactor in greater detail, by examining the evidence, supporting arguments and claims made in the safety documentation, building on the assessments already carried out for Steps 2 and 3, and to make a judgement on the adequacy of the Civil Engineering and External Hazards information contained within the PCSR and supporting documentation.

It is seldom possible, or necessary, to assess a safety case in its entirety, therefore sampling is used to limit the areas scrutinised, and to improve the overall efficiency of the assessment process. Sampling is done in a focused, targeted and structured manner with a view to revealing any topic-specific, or generic, weaknesses in the safety case. To identify the sampling for the Civil Engineering and External Hazards an assessment plan for Step 4 was set-out in advance.

My assessment has focused on:

- Resolution of issues arising from Step 3
- Load Schedule.
- Safety Classification of structures.
- Use of FE codes.
- Use of ETC-C.
- Containment design.
- Nuclear Island Design.
- Aircraft Shell design.
- Equipment qualification against hazards.
- Probabilistic Safety Analysis (PSA) and Seismic Margins Assessment (SMA).
- Decommissioning.

A number of items have been agreed with EDF and AREVA as being outside the scope of the Generic Design Assessment (GDA) process and hence have not been included in my assessment. These primarily relate to those aspects of the design which cannot be undertaken until a site has been chosen. These are discussed in more detail in the report. Those issues raised in Step 3 are inherently covered by the topics examined in Step 4.

From my assessment, I have concluded that:

<u>Load Schedule</u>

A review of the development of the loading schedule from identification of hazards through to their treatment within the hypothesis notes and within the ETC-C and finally their application in design

has been undertaken. I have concluded that whilst the hazard identification and screening is not fully transparent or documented, the outcomes are consistent with my expectations.

Load combinations are to be found in ETC-C, supplemented by event-based combinations. The ETC-C includes coincident hazard load combinations, such as from wind and snow. The "Event Based Approach" includes coincident hazards. This covers both physical phenomena inherent in the hazard (such as external flooding coincident with rainfall and high water table), and combinations of the hazard with independent internal or external initial conditions (such as the choice of temperature-dependent material properties for the earthquake loading condition). The "Event Based Approach" includes consequential hazards, such as fire following earthquake. The treatment of loading scenarios within the ETC-C has been the subject of considerable review, and this has culminated in the development of a greater degree of guidance in the UK companion document for the treatment of partial load factors. I am satisfied that the ETC-C and the UK companion document provide a clear basis for the application of design loads to the civil structures.

There are a number of hazards the magnitude or even presence of which cannot be defined until a site is identified. This has been recognised within the PCSR and it is clear that some further work will be required during the site specific design stage. In addition, a comprehensive review of the actual site specific hazards to ensure they are bounded by those values used in the generic design will be required.

A review by deep slice sampling of the design documentation for individual structures has revealed a number of concerns. These are typically linked to the lack of a clear audit trail for the building specific loading; information is often scattered across many documents, inconsistent treatment of loads; slabs designed for loads which are not subsequently explicitly checked for the supporting walls, and the dismissal of load cases without a complete justification. This will require resolution as part of the site specific design.

Safety Classification of Structures

The classification system is simple to understand, and for the major civil structures the rationale is clear. For some substructures within the main buildings the classification is not yet fully defined, however there is a commitment to provide an updated PCSR to address this concern.

The turbine hall is currently identified as a non safety classified structure, however it is a seismic class 2 structure, as its collapse may threaten some safety-related plant or equipment. However there is a remnant uncertainty over the safety categorisation of some plant within the turbine hall. This requires resolution ahead of a site specific design being undertaken.

Use of FE Codes

The process of designing the Nuclear Island structures has used a large range of analysis tools. This includes finite element codes, mesh generators, translation software, and design software. The codes used fall into two broad categories. Firstly, there are those which HSE has regulatory experience of and have examined on a number of occasions previously. The second type are those which we have no previous knowledge. The bulk of the codes fall into the latter category. A review of the codes based on their validation and verification status, development history and regulatory oversight has been undertaken, proportionate to their importance in the overall design process.

For the bulk of the codes, I am satisfied that they are suitable for the purposes they have been used for. The only major exception to this is code PROMISS3D which is used for the analysis of soil structure interaction. The methodology used (Boundary element method) is not considered suitable for the softer sites experienced in the UK unless some further validation of the results is undertaken via an alternative method. Some of the other codes have minor caveats on their future use.

Use of ETC-C

The use of the ETC-C as a design code has been examined in some detail. It has been developed as a specific code for the design of the civil engineering aspects of the EPR I have concluded that the ETC-C is essentially a set of design guidance notes that cannot be used without a wealth of supporting documentation, such as hypothesis notes, Eurocodes, national annexes, Euronorms and other design guidance documents, both EPR specific and other general guidance documents. There are a number of areas where the approach adopted has been questioned at a fundamental technical level. This is typically where ETC-C or the French National Annexe modifies the Eurocodes in a manner which is potentially non-conservative by comparison with either other extant nuclear standards or with UK regulatory expectations. As a result of these concerns, the ETC-C has undergone a series of modifications; typically around the treatment of creep, shrinkage, design of pre-stressing, shear design such that our expectations are met. The final position on ETC-C is that the 2010 AFCEN version in conjunction with a UK companion document has been proposed as the design code for the UK EPR.

The use of what is a Eurocode based approach for structures which have a requirement for higher than normal reliability such as nuclear structures is considered worthy of special consideration in the forward to the codes. As a result, considerable effort has been undertaken to satisfy ourselves that suitable levels of reliability can be provided by the ETC-C. This resulted in the development of specific studies on the achieved reliability of the containment structure against the design basis demands. These studies were initially found to be inadequate and have been recently re-issued, but not in sufficient time for inclusion in this report. A GDA Issue remains on completion of a satisfactory assessment of the reliability of the ETC-C, GDA Issue **GI-UKEPR-CE-05.** The assessment in this report is based on the initial submissions.

Part 1 of the latest version of the ETC-C, ETC-C AFCEN 2010 along with its UK companion document are not yet considered to be fully acceptable for the design of the civil structures for a UK EPR.

Part 2 of ETC-C, which is concerned with the implementation of the design has been examined in an earlier version, and has been found to be somewhat outdated and over reliant on French regulations which are inappropriate in the UK. The update of ETC-C AFCEN 2010 part 2 has not been examined in full as yet, due to the supporting documents being delivered late in the programme. Satisfactory review of part 2 will be required ahead of any decision to issue a DAC.

Part 3 of ETC-C is concerned with the full scale pressure testing of the EPR containment. In its current form, it is also considered to require further development.

A GDA Issue remains on completion of a satisfactory assessment of the ETC-C (parts 1-3), GDA Issue **GI-UKEPR-CE-02.**

A major assessment finding remains for the checking of structures designed against Revision B of ETC-C that will need to be revalidated against the AFCEN 2010 version in conjunction with the UK companion document.

Containment Design.

The containment structure has been subjected to considerable scrutiny. This is a result of its safety role in the management of design basis accidents and the use of bonded tendons in its design, which is novel for nuclear applications within the UK. All existing UK pre-stressed concrete nuclear containments and pressure vessels rely on the use of unbonded tendons in greased ducts.

Comparison of the grouted vs ungrouted approach is not necessarily a useful one, as both systems have their inherent advantages and disadvantages. I have therefore focussed in the assessment on ensuring that the design as presented has been developed to a sufficient degree that it meets the expectations of our Safety Assessment Principles (SAPs) rather than comparing it to what is a completely different design concept. I have satisfied myself in this process that there is nothing fundamental about the use of grouted in place tendons which is incompatible with the SAPs that would be avoided though the use of ungrouted tendons.

The basic design of the containment is relatively independent of the site it will be built upon. This is a result of the governing load cases being related to the build up of internal temperature and pressure following a design basis accident. The design of the containment itself has been undertaken by Coyne et Bellier (COB) and the steel liner and penetrations by Neyrpic Framatome Mécanique (NFM). COB have designed almost all pre-stressed concrete nuclear containment structures currently operating in France and have considerable experience in this area.

The design is against the requirements of the ETC-C, and considerable effort has been spent in assessing the methodology within that code as well as the reliability of the code for the design of this type of structure. A range of computer codes have been used for analysis of the containment structure, the majority of which are in-house codes rather than more broadly available commercial codes. A detailed assessment of the principles of these codes, their development, regulatory acceptance and verification and validation has been undertaken. They have been found to be suitable for this purpose.

The detailed design of the containment has evolved over many years, and in order to achieve a complete understanding, I have had to examine documentation developed as part of the Basic Design Optimisation Phase (BDOP) where key decisions were made over the design philosophy for the EPR containment. It should be noted that the EPR is the first double walled lined containment developed by EDF. Previous double wall containments have been unlined, and in this respect there is a degree of novelty in the design. Previous experience from EDF in the behaviour of containment structures, particularly in terms of creep and shrinkage has impacted on their design methodology. In addition, research work undertaken at the Maeva mock up has provided valuable benchmarking of the predicted behaviours.

Inherent in the choice of grouted in place tendons is the expectation that there will be no significant degradation of the tendons through life. I have therefore focussed on the implementation of the grouting, grouting trials, the level of redundancy in the containment and the monitoring systems for the containment. I have come to the conclusion that the containment monitoring undertaken is capable of detecting a degradation in the performance of the pre-stressing system well ahead of the degradation reaching a stage where it compromises the design basis. There are still a number of assessment findings related to grout composition, trials and some aspects of monitoring which require addressing on a site specific basis.

The information provided on the containment behaviour for beyond design basis accident loads has been found to be insufficient. In addition, there remain a series of unresolved concerns over the analysis approach used for the containment. Two GDA Issues have been raised on these points **GI-UKEPR-CE-03 and -04.**

Nuclear Island Design

The design of the Nuclear Island structures has been addressed at different levels in the GDA process. The containment structure and the Air Plane Crash (APC) shell have been examined in great detail. The Safety Auxiliaries Building (SAB) and Nuclear Auxiliaries Building (NAB) have been examined at a fairly high level, on the basis that they will undergo substantial redesign in parts for application in the UK, and that they are both standard types of structures, the design of which should be relatively routine. The fuel building and the inner containment structures have been examined at an intermediate level of detail, as they are unlikely to have significant redesign, but do have some unique features which require careful design.

The hypothesis documents for all the Nuclear Island structures have been examined and found to be heavily biased towards the Flamanville 3 project and will need modification to apply to a UK EPR. In addition, there are a number of changes and additions which are required to ensure they are suitable for use in the UK. A GDA Issue remains on production of satisfactory hypothesis notes for the Nuclear Island structures, GDA Issue **GI-UKEPR-CE-01.**

The fuel building design has been examined in some detail, particularly the fuel pool itself. The detailed design of the pool liner has not been undertaken at a generic level, and hence the assessment has been against the ETC-C guidance and support documentation and methodologies. This has confirmed that the liner design process is robust, and considered acceptable. The design of the concrete portion of all steel lined concrete pools which have a permanent and potentially contaminated fluid will need to be confirmed as adequate against the requirements of BS-EN 1992 part 3 (Tightness class 1).

The containment internal structures have generally been found to be reasonably well designed. There are some residual concerns however which will need to be addressed by any future licensee, particularly in the area of the Polar Crane supports.

The methodology for seismic analysis of the structures has been the subject of extensive discussions, and a series of methodology documents have been produced by EDF and AREVA during Step 4. A GDA Issue remains on completion of a satisfactory assessment of the seismic methodology, GDA Issue **GI-UKEPR-CE-06.**

Aircraft Shell Design

The approach used for the protection of safety critical structures sytems and components against the threat from aircraft crash has been examined, including a review of the codes and standards used. In particular, the design of the APC shell for the nuclear island has been assessed in some detail. I am satisfied that the design of the APC shell is satisfactory to withstand impacts from military and commercial aircraft such that essential safety functions can be maintained.

The assessment of the detailed design of doors and openings in the shell, and more broadly in the Nuclear Island has not been possible during GDA due to difficulties over the late delivery of information and on the exchange of sensitive information. The basic principles have however been agreed.

Equipment Qualification Against Hazards

At this stage, a limited review of the qualification of plant and equipment has been possible. This has been restricted to a review of overall methodologies and approaches rather than the details of individual qualifications. This has shown that the methodologies are broadly acceptable, however there will be a requirement to examine this aspect further at the site specific stage.

PSA and SMA

A Probabilistic Safety Analysis (PSA) based seismic margins assessment has been undertaken by EDF and AREVA which is consistent with the methods described and advocated in EPRI documents, which are in line with my expectations. The methodology utilised by EDF and AREVA to determine the High Confidence of Low Probability of Failure (HCLPF) values is the probabilistic fragility analysis method which is considered acceptable. The intention of EDF and AREVA is not to derive realistic fragilities for use in a seismic PSA, but to demonstrate that their target has been met in a seismic margins assessment. A major shortcoming is the availability of information at the GDA stage, on which the fragility evaluation can be performed. There is therefore a finding to review this position at the site specific level, once more detailed information is available.

Decommissioning

A review of the construction techniques and design approach for the Nuclear Island structures has shown that they can be decommissioned using techniques currently available. Decontamination of the fuel pool structure may require some degree of gradual demolition/ excavation of buried components in the walls, however this is readily achievable if somewhat involved. Careful consideration will be required during demolition of the inner containment structure as a result of the large amount of entrained energy within the pre-stressing tendons.

Conclusions

In some areas there has been a lack of detailed information which has limited the extent of my assessment. As a result, HSE-ND will need additional information to underpin my conclusion and these are identified as Assessment Findings to be carried forward as normal regulatory business. These are listed in Annex 1.

Some of the observations identified within this report are of particular significance and will require resolution before HSE would agree to the commencement of nuclear safety-related construction of a UK EPR reactor in the UK. These are identified in this report as GDA Issues are listed in Annex 2. In summary these relate to:

- Hypothesis notes for the Nuclear Island (NI) Structures **GI-UKEPR-CE-01.**
- Use of ETC-C and the reliability it assures **GI-UKEPR-CE-02 and 05.**
- Detailed methodologies for the treatment of SSI and seismic analysis **GI-UKEPR-CE-06.**
- Beyond Design Basis behaviour of the Inner Containment **GI-UKEPR-CE-03 and 04.**

Overall, based on the sample undertaken in accordance with ND procedures, I am broadly satisfied that the claims, arguments and evidence laid down within the PCSR and supporting documentation submitted as part of the GDA process present an adequate safety case for the generic UK EPR reactor design. The UK EPR reactor is therefore suitable for construction in the UK, subject to satisfactory progression and resolution of GDA Issues to be addressed during the forward programme for this reactor and assessment of additional information that becomes available as the GDA Design Reference is supplemented with additional details on a site-by-site basis.

LIST OF ABBREVIATIONS

AE	Acoustic Emission
AFCEN	Association Française pour les règles de conception et de construction des matériels des Chaudières ÉlectroNucléaires
AFNOR	Association française de Normalisation
ALARP	As Low As Reasonably Practicable
APC	Air Plane Crash
ASCE	American Society of Civil Engineers
ASN	Autorité de Sûreté Nucléaire (French nuclear safety authority)
BEM	Boundary Element Method
BMS	(Nuclear Directorate) Business Management System
BS	British Standard
BSL	Basic Safety Level (in SAPs)
BSO	Basic Safety Objective (in SAPs)
CEIDRE	Centre d'Expertise et d'Inspection dans le Domaine de la Réalisation et de l'Exploitation
CNEN	Centre National Equipment Nucléaire
COB	Coyne et Bellier
CRDM	Control Rod Drive Mechanism
DBE	Design Basis Earthquake
DECC	Department of Energy and Climate Change
DfT	Department for Transport
EDF and AREVA	Electricité de France SA and AREVA NP SAS
EMI	Electro-Magnetic Interference
ETC-C	EPR Technical Code - Civil
FA3	Flamanville 3 Nuclear Power Plant
FEA	Finite Element Analysis
GACM	German Association for Computational Mechanics
GDA	Generic Design Assessment
GQAS	General Quality Assurance Specifications
HCLPF	High Confidence Low Probability of Failure
HSE	The Health and Safety Executive
IABSE	International Association for Bridge and Structural Engineering
IACM	International Association for Computational Mechanics

LIST OF ABBREVIATIONS

IAEA	The International Atomic Energy Agency
JRC	Joint Research Centre
LOCA	Loss of Cooling Accident
MCR	Main Control Room
MDEP	Multinational Design Evaluation Programme
MSQA	Management Systems and Quality Assurance
NAB	Nuclear Auxiliaries Building
NAFEMS	National Agency for Finite Element Methods
NCB	Non Classified Building
ND	The (HSE) Nuclear Directorate
NDA	Nuclear Design Associates
NEP	Non Exceedance Probability
NFM	Neyrpic Framatome Mécanique
NPP	Nuclear Power Plant
OCNS	Office for Civil Nuclear Security
OJEU	Official Journal of the European Union
OL3	Olkiluoto 3
PCER	Pre-construction Environment Report
PCSR	Pre-construction Safety Report
pga	Peak Ground Acceleration
PID	Project Initiation Document
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
RFS	Règles Fondamentales de Sûreté
RGP	Relevant Good Practice
RI	Regulatory Issue
RIA	Regulatory Issue Action
RO	Regulatory Observation
ROA	Regulatory Observation Action
RPV	Reactor Pressure Vessel
SAB	Safety Auxiliaries Building
SAP	Safety Assessment Principles
SASSI	Seismic Analysis of Soil Structure Interaction
SDM	System Design Manuals

LIST OF ABBREVIATIONS

SEL	Seismic Equipment List
SEPTEN	Service Etudes et Projets Thermiques et Nucléaires
SLS	Serviceability Limit State
SFAIRP	So Far As Is Reasonably Practicable
SMA	Seismic Margins Assessment
SME	Seismic Margin Earthquake
SML	Submission Master List
SQEP	Suitably Qualified and Experienced Personnel
SSC	System, Structure and Component
SSER	Safety, Security and Environmental Report
STUK	The Finish Nuclear Safety Authority
TAG	(Nuclear Directorate) Technical Assessment Guide
TQ	Technical Query
TSC	Technical Support Contractor
UHS	Uniform Hazard Spectrum
ULS	Ultimate Limit State
US NRC	Nuclear Regulatory Commission (United States of America)
WENRA	Western Europe Nuclear Regulators' Association

Building Abbreviations

HD	Diesel Building
HK	Fuel Building
HL	Safety Auxiliaries Building
HN	Nuclear Auxiliaries Building
HR	Reactor Building

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1 INTRODUCTION

1 This report presents the findings of the Step 4 Civil Engineering and External Hazards assessment of the November 2009 UK EPR reactor PCSR (Ref. EA1) and supporting documentation provided by EDF and AREVA under the Health and Safety Executive's (HSE) Generic Design Assessment (GDA) process. Assessment was undertaken of the PCSR and the supporting evidentiary information derived from the Submission Master List (Ref. EA252). The approach taken was to assess the principle submission, i.e. the PCSR, and then undertake assessment of the relevant documentation sourced from the Submission Master List on a sampling basis in accordance with the requirements of the Nuclear Directorate's (ND) Business Management System (BMS) procedure AST/001 (Ref. ND2). The Safety Assessment Principles (SAP) (Ref. ND5) have been used as the basis for this assessment. Ultimately, the goal of assessment is to reach an independent and informed judgment on the adequacy of a nuclear safety case.

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

3 A number of items have been agreed with EDF and AREVA as being outside the scope of the GDA process and hence have not been included in this assessment.

2 NUCLEAR DIRECTORATE'S ASSESSMENT STRATEGY FOR CIVIL ENGINEERING AND EXTERNAL HAZARDS

4 The intended assessment strategy for Step 4 for the Civil Engineering and External Hazards topic area was set out in an assessment plan (Ref. ND11) that identified the intended scope of the assessment and the standards and criteria that would be applied. This is summarised below:

5 This is the third major report in GDA on the assessment of the civil engineering and external hazards aspects of the design of the UK EPR.

6 The original intention for GDA was that Step 3 should be an assessment of the arguments provided to support the claims assessed in Step 2. Step 4 would then examine the evidence to support the arguments. It is difficult in the areas of civil engineering and external hazards to separate out the arguments and evidence in a meaningful way. An approach of examining the principles used in the design within Step 3 and their application in Step 4 has been adopted wherever possible.

7 The design of the civil structures has been undertaken using non-UK design codes, supported by the use of finite element codes which are typically unfamiliar in the UK. There has therefore been a considerable learning curve during Steps 3 and 4.

8 The volume of information to examine has led to extensive use of technical support contractors to provide expertise across a wide range of areas.

9 A process of regular meetings with EDF and AREVA to discuss technical issues, monthly teleconferences and the use of the TQ and RO process has ensured that there has been continuous dialogue throughout Step 3 and 4.

10 The reference design is that adopted for Flamanville 3 EPR. However it is recognised that some of the structures are site specific, and can only be considered in detail once a site has been selected and the necessary studies undertaken.

2.1 Assessment Plan

11 An assessment plan was developed at the start of Step 4 and can be seen in Ref. ND11.

12 The plan focussed on the following areas:

- Resolution of issues arising from Step 3.
- Load Schedule.
- Safety Classification of structures.
- Use of Finite Element (FE) codes.
- Use of ETC-C.
- Containment design.
- Nuclear Island Design.
- Aircraft Shell design.
- Equipment qualification against hazards.
- Probabilistic Safety Analysis (PSA) and Seismic Margins Assessment (SMA).
- Decommissioning.

13 The plan made certain assumptions over the delivery of information from EDF and AREVA and inherent assumptions over the acceptability of that information to ND. Some of these assumptions have proved to be correct and others less so. This is documented in the individual sections of this report.

14 During the course of the assessment, the scope of the topics covered within GDA has been developed and is now commonly agreed with EDF and AREVA.

2.2 Standards and Criteria

15 During Step 4 our assessment of the proposed design is against those principles in the HSE Safety Assessment Principles for Nuclear Facilities (SAP), that are deemed relevant to system design aspects (see guidance below).

16 With regard to the Western European Nuclear Regulators' Association (WENRA) Reference Levels (Ref. OD1), the foreword to the new SAPs notes that *"In the UK, the (WENRA) reference levels will be secured using a combination of .... SAPs"*, hence assessment against the SAPs is considered sufficient. However, I have considered whether the SAPs include the key WENRA principles relevant to civil engineering and external hazards.

17 The SAPs represent HSE's view of good practice and HSE would expect modern facilities to have no difficulty in satisfying their overall intent. Meeting relevant good practice is an essential part of demonstrating adequate safety and in satisfying the As Low As Reasonably Practicable (ALARP) principles. In defining relevant good practice, the GDA Guidance, states, "…what may be regarded as good practice and what is reasonably practicable might be found in the design of reactors currently operating or under construction or licensing elsewhere in the world, including the Sizewell B design in the UK". The precedents set by Sizewell B will be used, amongst others, as a reference point for establishing relevant good practice in the UK.

18 The use of the SAPs is supplemented, as appropriate, with NII Technical Assessment Guides (TAG). The TAGs (Refs ND6 and ND7) provide further interpretation of the SAPs and guidance in their application. An important part of the assessment process is determining whether appropriate modern standards have been used by EDF and AREVA (SAP ECS.3 'Standards'). Consequently, particular attention has been paid to such claims (e.g. has the RP claimed adequate standards selected and applied).

19 The scope of the principles in the SAPs is extensive. They cover all nuclear facilities, i.e. nuclear power plants, fuel cycle facilities, including radioactive waste management, and cover all phases of the facility life-cycle, i.e. design, construction, commissioning, operation and decommissioning. Consequently, not all of the principles in the SAPs apply to a review of the fundamental safety claims included in the PCSR information for a nuclear power plant. In determining the appropriate SAP coverage (selection and sampling), the following has been considered:

- Has the RP claimed coverage of all SAPs and provided adequate information in the safety case for the arguments to support the fundamental claims?
- The list of key SAPs relevant to each topic area.
- I have been selective in my confirmation of SAPs coverage in Step 4, e.g. through confirmation of credible claims and supporting arguments for the key SAPs in each topic area.
- Judgment has been used in selecting those SAPs for assessment at Step 4 and the level of detail to which the assessment will be taken.

20 In making judgments on whether a SAP was relevant to 'fundamental design aspects' (i.e. should be included in the list of key SAPs) and needs, therefore, to be considered during Step 4, the following factors were considered:

- The SAP addresses the selection of modern design standards.
- The SAP significantly addresses plant architecture and layout.
- The WENRA Reference Levels support the selection of the SAP.

21 The International Atomic Energy Agency (IAEA) document 'Safety of Nuclear Power Plants: Design – Requirements – NS-R-1' (Ref. OD2) supports the selection of the SAPs.

22 In order to ensure an adequate set of SAPs for Step 3 and 4 a further review of the WENRA Reference Levels (Ref. OD1) and the IAEA Nuclear Power Plant (NPP) Design Requirements (Ref. OD2) was undertaken. The SAPs selected for assessment of claims and arguments during Step 3 and 4 are shown in Annex 2 of Ref. ND11 where they are ordered under assessment topic areas. This is repeated as Table 5 in this document.

2.3 Assessment Scope

23 The assessment scope for Civil Engineering and External Hazards is broken down into two basic areas, physical structures and methodologies. The sections below provide further details on those items which are considered as included in GDA and those which are not.

24 All documentation that has been provided by EDF and AREVA is listed on the Submission Master List (SML) (Ref. EA252). The SML is split into four levels of documentation and lists all the documents and records which support the GDA submission and have been included in our assessment. The four levels are described below:

- Level 1 – safety, security and environmental report.
- Level 2 – documents referenced in the safety, security and environment reports, such as the design reference and Stage 1 System Design Manuals (SDM).
- Level 3 – submission supporting documents provided to the Regulator for assessment, supporting TQ/ RO/ RI and meeting action responses.
- Level 4 – project or site specific documents for information only, which have been used in the assessment to achieve regulatory confidence that the design outline in GDA can be developed to the construction stage, such as Stage 2 SDMs.

The Level 4 documents are project or site specific documents that are provided for information only. They have been used in the assessment to achieve regulatory confidence that the design outline in GDA can be developed to the construction stage, for example, the Stage 2 SDMs. This Level 4 information will not be included in the scope of the GDA submission for any interim Design Acceptance Confirmation (iDAC) / Design Acceptance Confirmation (DAC) we may issue.

2.3.1 Physical Structures - Inclusions

25 The key physical areas of inclusion in GDA are:

- Nuclear Island including:
  i) Safety Auxiliaries Building (SAB) 1 to 4.

ii) Inner Containment structure.

iii) Fuel Building.

iv) Aircraft Crash Shell.

- Nuclear Auxiliaries Building.

2.3.2 Physical Structures - Exclusions

26 At this stage in GDA the detailed design of the following elements has not been reviewed.

- Nuclear Island Foundation (See notes below)
- Nuclear Auxiliaries Building Foundation (See notes below)
- Diesel Building (See notes below)
- Waste Treatment Building (Design not developed)
- Cooling Water Pumphouse (Site specific)
- Cooling Water Intake Structures (Site Specific)
- Ancillary Buildings (Site Specific)
- Ancillary services and
- Structures, i.e. tanks, service trenches (Site Specific)
- Sea walls (Site Specific)
- Nuclear Auxiliaries Building (NAB) Chimney (See notes below)
- Access Tower

The foundation design for the Nuclear Island and the Nuclear Auxiliaries Building has been completed for the reference design (FA3), however this is not necessarily representative of all potential UK sites. Following discussion with EDF and AREVA agreement was reached on the scope of the foundation review in GDA. This is formally documented in Ref. EA11.

2.4 Methodologies - Inclusions Civil Engineering

27 The following sections provide a more detailed description of the documents considered for each of the key structures. These methodologies are typically within “hypothesis notes”.

2.5 Civil Engineering Works – General

28 The bulk of the design of the civil engineering works for the reference design from Flamanville 3 has been undertaken using ETC-C Rev B (2006) part 1, (Ref. EA3). For the UK, a revised version of ETC-C is proposed, ETC-C AFCEN 2010 (Ref. EA74). For the purposes of GDA, while the ETC-C Rev B was included in the assessment process, it is the later ETC-C AFCEN document that is being assessed as being the GDA design code, along with its UK companion document (Ref. EA73). It should be noted that some of the deep slice evaluations of structures have examined designs undertaken to ETC-C Rev B. The shift in proposed design code is reflected in the assessments and subsequent assessment findings however.

29 The ETC-C Part 2 provides an overview of the construction requirements. There has been extensive discussion with EDF and AREVA over the scope of the ETC-C part 2 in GDA.

30 ETC-C Part 2 and clarification of the construction rules for UK EPR are considered as part of the GDA Scope. Detailed assessment of the following sections has been undertaken:

- Concrete structures (i.e. ETC-C sections 2.2 to 2.5 – Concretes, Facing and forms, Reinforcement for reinforced concrete, Pre-stressing system)
- Metal structures:
  - Section 2.7 (Leak-tight metal parts of containment);
  - Section 2.9 (Watertight and tank metal liner); and
  - Section 2.10 (Structural Steelworks).
- Associated tolerances (extract of Section 2.13) linked to the previous rules.

31 Part 3 of the ETC-C, which covers full scale pressure testing and monitoring of the containment, is also included in GDA.

32 The following documents provide the methodologies and approaches which have been examined within GDA:

- ECEIG021405 Rev H1 — General Hypothesis Note for Civil Engineering Design of Nuclear Island Buildings (Ref. EA12).
- ECEIG050051 Rev L1 — Note on civil engineering standards (Ref. EA13).
- ECEIG051339 Rev B1 — Note on the general philosophy for the metal frames (Ref. EA14).
- ECEIG060543 Rev C1 — CDC requirements and design of the pools/ponds (Ref. EA15).
- ENGSGC080086 Rev B1 — Methodology for consideration of Shrinkage for EPR Structures. (Ref. EA16).

33 The cooling pool design for the GDA has been examined at a principles level. The approach in the ETC-C code is primarily based on physical testing of a particular design and limits have been placed on the strain/ displacement. The ETC-C is very limited in its extent, and through the assessment process we have requested further details from EDF and AREVA on the application of these rules. The design of the pool for Flamanville was undertaken by Principia (www.principia.fr), who are a subsidiary of AREVA. The contract for the design of the pond liner however was let by Bouygues, the main civils contractor for Flamanville as part of the design and construction works. The design therefore is entirely Flamanville specific and is within the control of EDF via its main contractor to its subcontractor. Whilst a meeting to discuss the detailed design has been held, and is discussed further in Section 4.3.5.3.3, the detailed design of the liners is out with the scope of GDA.

34 The scope of GDA as far as cooling pool liner design is concerned is limited to the methodologies contained within Refs EA15 and EA86 and the ETC-C AFCEN.

35 Document ECEIG061031 Rev A (Ref. EA17) General Principles for the design of Drains and Sumps has not been examined in GDA.

2.5.1.1 Nuclear Island – General

36 The following documents provide the methodologies and approaches which have been examined within GDA.

- SFL-EYRC-0030027 C1 — Assumptions Report Re EPR FA3 Common Foundation Raft Design (Ref. EA18).
- ECEIG021405 Rev H1 — General Hypotheses Note for Civil Engineering Design of Nuclear Island Buildings (Ref. EA12).
- ENGSGC100140 Rev B — Common foundation raft - GDA Scope (Ref. EA11).
- 10439-NT-28B01-0101 F1 — Nuclear Island 3D Overall model Hypothesis and Methodology note (Ref. EA20).
- 10439-NT-28B01-0103 B1 — Nuclear Island 3D Overview model - Determination and Location of Equipment Masses and Operating Loads (Ref. EA21).
- 10439-NT-28B01-0104 D1 — Nuclear Island 3D Model used to determine soil structure interaction parameters (Ref. EA22).
- 10439-NT-28B01-0105 D1 — Nuclear Island 3D Model Modal analysis - calculation of spectral responses (Ref. EA23).
- 10439-NT-28B01-0106 C1 — Nuclear Island Overall 3D Model Results of basic seismic load combinations- displacements - accelerations – reactions (Ref. EA24).
- 10439-NT-28B01-0107 C1 — Nuclear Island 3D General Arrangement Model- Description of the static load cases – Reactions Forces Displacements (Ref. EA25).
- 10439-NT-28B01-0108 C1 — Nuclear Island 3D Overview Common Raft Foundation Raft Reinforcement for static and dynamic load combinations (Ref. EA26).
- 10439-NT-28B01-0109 C1 — Nuclear Island 3D Overview Evaluation of Common Foundation Raft and Containment Internals Raft Lifting in the Event of Earthquake (Ref. EA27).
- 10439-NT-28B01-0114 A1 — Nuclear Island 3D Overall Model Sensitivity Calculation with Ground Structure Interaction Parameters – Validation of static and dynamic load cases (Ref. EA28).

37 The scope of the seismic analysis considered for GDA needs further explanation. There are two sets of analysis. The first set is used for deriving secondary response spectra within the buildings for equipment qualification. This analysis covers a range of soil types and provides an envelope response for future use. The second analysis is based only on the Flamanville 3 site conditions, and is used for deriving loads in structures for design of elements. Both analysis sets are included in the review, however there are different limitations on their future use for a design to be built in the UK. For equipment qualification, it may be possible to undertake some limited comparison to show that the response on a UK site would be within the envelope of soil conditions considered. For the second case however there would be a more involved re-analysis process to confirm the loads used in the FA3 design enveloped those for a UK site. The alternative would

be to generate the structural loads again from a UK specific analysis of the Nuclear Island.

2.5.1.2 Reactor Building

38 The following documents provide the methodologies and approaches which have been examined within GDA.

- ECEIG0001089 Rev C1 — Reactor Building Specifications (Ref. EA29).

During the Step 4 assessment, this document has been declared out of the scope of GDA, however our comments have been included for completeness.

2.5.1.3 Inner Containment

39 The following documents provide the methodologies and approaches which have been examined within GDA

- ECEIG102044 Rev B — Inner Containment Wall Detailed Design Report (Ref. EA30).
- SFL EYRC 0030018 Rev E1 — Hypothesis Note on Inner Containment Wall Fitted with Steel Skin Inside Reactor (Ref. EA31).
- 11815 28B03 NT 003 Rev E1 — Reactor building – Preliminary studies – Inner containment wall. Inner containment wall dimensioning hypothesis report (Ref. EA32).
- 11815 28B03 NT 007 Rev C — Reactor building – Pre-stressed concrete inner containment wall detailed design definition of basic static load. (Ref. EA33).
- 892 CD 01001 Rev C — Inner containment wall. Steel liner design. Definition and description of design assumptions (Ref. EA34).
- 892 CD 01026 Rev C1 — Inner containment wall. Analysis of the steel liner. Design hypothesis report on the 273 to 1422 mm diameter penetration sleeves (Ref. EA35).

2.5.1.4 Inner Containment Internal Structures

40 The following documents provide the methodologies and approaches which have been examined within GDA

- SFL EYRC 003022 Rev F1 — Hypothesis Note on Reactor Building Containment Internals. (Ref. EA36).
- 11787-YR1221-NT-28B01-0001 Rev D1 — Reactor Building - Containment Internals - Assumptions and Methodology Note (Ref. EA37).

2.5.1.5 Fuel Building

41 The following documents provide the methodologies and approaches which have been examined within GDA

- ECEIG99070 Rev B1 — Fuel Building Specifications (Ref. EA38)

During the Step 4 assessment, this document has been declared out of the scope of GDA, however our comments have been included for completeness.

- SFL EYRC 0030017 Rev F1 — Note design assumptions civil engineering building fuel FA3 (Ref. EA39).
- 10439-NT-28B01-301 Rev E1 — Fuel Building – Assumptions and Methodology note (Ref. EA40).

42 The NAB chimney is located above the Fuel Building on the APC shell. The following documents provide the methodologies and approaches which have been examined within GDA.

- ECEIG080033 — Note on the philosophy of the DVN chimney (Ref. EA41).
- EYRC/2009/FR/0088 Issue C1 — YR1221 Hypothesis note for the Bouygues design of the DWN [NABVS] stack on the fuel building roof (Ref. EA42).

43 The detailed design of the NAB chimney has not been considered in GDA as there are site specific influences on the design such as wind speed and local topography.

2.5.1.6 Safety Auxiliaries Buildings

44 The following documents provide the methodologies and approaches which have been examined within GDA

- SFL-EZC-00-8002 Rev C — Hypothesis Note for the Safety Auxiliary Building Divisions 2 and 3 (Ref. EA43).
- SFL-EZC-00-8001 Rev C — Hypothesis Note for the Safety Auxiliary Building Divisions 1 and 4 (Ref. EA44).
- 10439-NT-28B01-0401 D — SAB Div 2 and 3 - Hypotheses notes (Ref. EA45).
- 11787-YR1221-NT-28B01-0501 D — Safety Auxiliary Building - Divisions 1 and 4 - Note of hypotheses and methodology (Ref. EA46).

45 The Main Control Room (MCR) is located in the SAB 2 building. It is essentially an independent structure within the SAB, but, isolated from the SAB by a series of “Gerb” springs the design for FA3 is not fully completed. Following discussions with EDF and AREVA, the following was agreed

Within GDA scope:

- Design methodology
- Main load cases, load combinations and assumptions.
- Material characteristics.

Outside GDA scope:

- Detailed design

- Fitting layout.

46 The following documents provide the methodologies and approaches which have been examined within GDA

- SFL-EEZ-00-8-016 Rev B — Concept Design of the MCR for EPR France Specific Arrangement (Ref. EA47).
- EDF document reference ECEIG060426 — EPR – MCR Technical Review – Seismic Behaviour, 12 April 2006 (Ref. EA238).
- Tobias Richter, Transient Time History Analysis and Optimisation of the Dynamic Behaviour of a Command Room Isolated on A-seismic Bearing Pads, Diploma Thesis, Faculty of Civil Engineering, Institute of Structural Analysis and Structural Dynamics, Dresden University of Technology, in Cooperation with AREVA, September 2006 (Ref. EA239)

2.5.1.7 Turbine Hall

47 Document ETDO1G/070090B1 BPE "EPR Flamanville 3 - General Hypothesis Note for Turbine Hall Design" (Ref. EA72) has been examined in GDA.

48 The detailed design of the turbine hall has not been considered in GDA as there are site and operator specific influences on the design which require consideration.

2.5.1.8 Diesel Building

49 The following documents provide the methodologies and approaches which have been examined within GDA

- ECEIG0000756 Rev C1 — Technical Specification for Diesel Buildings (Ref. EA48).

During the Step 4 assessment, this document has been declared out of the scope of GDA, however our comments have been included for completeness.

- SFL EYRC 0030020 Rev E1 — Note on Civil Engineering Design of the EPR FA3 Diesel Building. Ref. EA49).
- 10439-NT-28B01-501 Rev D1 — Diesel Building Hypothesis and Methodology Report (Ref. EA50).

50 The detailed design of the Diesel Building has not been considered in GDA

2.5.1.9 NAB Building

51 The following documents provide the methodologies and approaches which have been examined within GDA.

- ECEIG0000333 Rev E — Specifications for Nuclear Auxiliary Building (Ref. EA51).

During the Step 4 assessment, this document has been declared out of the scope of GDA, however our comments have been included for completeness.

- SFL EZC 00 8003 C — Hypothesis note for the Nuclear Auxiliary Building (Ref. EA52).
- 11788-YR1222-NT-28B01-0001-D — Note of Hypotheses and Methodology (Ref. EA53).

2.5.1.10 Aircraft Protection Shell

52 The detailed evaluation of the aircraft protection shell is out with this document, other than to present the overall conclusions.

53 The areas which are within the scope of GDA are as follows.

- Overall methodology for the treatment of accidental and non-accidental aircraft crash
- Design of the aircraft protection shell for the Nuclear Island
- Methodology for the design of penetrations and openings in aircraft protection structures.

2.5.1.11 Computer Codes

54 In the work undertaken to review the design of the Civil Structures there has been a considerable effort to understand the computer codes used for analysis and design of structures. There is a complex interrelationship between the different codes, which has been examined in some detail. In addition, there are assorted manipulation, translation and code checking software which has been used in the design of the UK EPR.

55 Table 1 below identifies all the computer codes and other software examined as part of the GDA assessment.

Table 1: Summary of Computer Codes Examined in GDA

Code	Type of Code	Used on
ASTER	Finite Element	NI Structures
COBEF	Finite Element	NAB SAB 1 and 4
HERCULE	Finite Element	SAB 2,3 Fuel, Diesel
ANSYS	Finite Element	Containment
SYSTUS	Finite Element	Containment Liner
NASTRAN	Finite Element	NAB Building
SOFISTIK	Finite Element	Dynamic Behaviour of Structures
EUROPLEXUS	Finite Element	Dynamic Behaviour of Structures
MISS 3D/PROMISS	BEM Finite Element	Ground Model
ASTHER	Translator	SAB 2,3 Fuel, Diesel
HERAST	Translator	SAB 2,3 Fuel, Diesel
PRECONT	GEOMETRIC	Containment
FERRAIL	Code Checker	SAB 1,4,NAB

Code	Type of Code	Used on
HFERCOQ	Code Checker	SAB 2,3 Fuel, Diesel
SIGNSOL	Load Manipulation	SAB 2,3 Fuel, Diesel
CASTEM	MESH GEN	NI Structures
HER 2 COB	Translator	NI Structures
COBEF to HERCULE	Translator	SAB 1,4
COBEF to NASTRAN	Translator	NAB

2.5.1.12 Decommissioning

56 The detailed decommissioning arrangements for the EPR civil structures have not been examined in GDA. An assessment of the practicability of dismantling the structures has been made. Further details on decommissioning can be found in the Radwaste and decommissioning topic report.

2.5.2 Methodologies - Exclusions Civil Engineering

57 There are a number of areas which are not covered in GDA. These can be summarised as.

- Maintenance of Civil Structures.
- Geotechnical Investigation and Design (ETC-C-Part-2.1)

2.5.3 Methodologies - Inclusions External Hazards

58 The methodology for the identification and screening of external hazards that have been considered in the design of the EPR are included in GDA This is presented in the following documents:

- ENSN040070 Rev A1 Presentation of Approach for incorporating hazards in the EPR project (Ref. EA2).
- ENSNEA080058-A EPR – External Hazards – Inventory of Combined Events with Internal Faults and/or other (Internal and External) Hazards taken into account in Design (Ref. EA10).

59 The design of the Nuclear Island structures and the NAB against seismic loading is within the scope of GDA.

60 The design of the Nuclear Island structures against aircraft impact is within the scope of GDA.

61 The derivation of seismic fragilities has been assessed within GDA.

62 The approach to the qualification of equipment against external hazards has been examined in GDA at a principles level. The following documents have been examined.

- BTR 91 C 112 Rev 00 Equipment earthquake resistance test – Generic provisions for the biaxial time history test (Ref. EA148).
- ECEF0000837 Rev E1 EPR equipment classification list (Ref. EA149).

- ECEF040759 Rev D1 EPR - Preliminary list of plant functions to be qualified for accident ambient conditions (Ref. EA150).

2.5.4 Methodologies - Exclusions External Hazards

63 The derivation of site specific or UK envelope external hazards is excluded from GDA.

64 Confirmation that the EPR design basis load-cases for external hazards envelope UK demands is excluded from GDA. A broad comparison of the design basis events used in the reference design against expected UK values has been done to give some confidence that the reference design would be suitable for UK application.

65 The methodologies for the derivation of external hazard levels for specific sites has not been considered in GDA.

2.5.5 Findings from GDA Step 3

66 The following paragraphs are a summary of the key findings taken from the Step 3 report.

67 The analysis and design of the civil structures has been undertaken using primarily French or European codes and standards, about which we had little knowledge at the start of GDA. There has been a considerable learning curve therefore before substantive assessment could commence.

68 The development of the design basis load cases within the documentation has taken some unravelling, however I am broadly content with the final outcome of the process used. Some further sampling of the detailed design approach will be undertaken in Step 4.

69 The analysis codes used to predict the behaviour of the structures during extreme loading scenarios are in the process of being assessed. The work undertaken thus far has indicated that the bulk of the codes will be found to be suitable for their chosen application, however further sampling of the application of these codes in Step 4 will be undertaken.

70 The use of the ETC-C as a design code has been examined in some detail. It has been developed as a specific code for the design of the civil engineering aspects of the EPR, We have concluded that the ETC-C is an in-house set of design guidance notes that cannot be used without a wealth of supporting documentation. There are a number of areas where the approach adopted is being questioned at a fundamental technical level. This is typically where ETC-C or the French National Annexe modifies the Eurocodes in a manner which is potentially non-conservative by comparison with either other extant nuclear standards or with UK regulatory expectations. The use of Eurocodes for structures which have a requirement for higher than normal reliability such as nuclear structures is considered worthy of special consideration in the forward to the codes. As a result, considerable effort is being undertaken to satisfy ourselves that suitable levels of reliability can be provided by the ETC-C. In addition, there are a number of references to superseded codes and practices, or a lack of rigour in the approach to be adopted in key areas. Clearly, the manner in which the codes have been applied is key to the acceptability of the design. This will be explored in Step 4.

71 The inner containment has been examined in some detail for two key reasons; firstly the safety demands placed upon it and secondly the use of unbonded pre-stressing tendons, a novel approach in the UK for nuclear applications. The initial responses from EDF and AREVA to our queries were disappointing; however the more recent exchanges have been more promising. There is still a considerable amount of justification to be

undertaken to convince me that the design approach is consistent with our regulatory expectations, however it is considered that this is practicable.

72 Progress on the assessment of the aircraft protection shell has been hampered by difficulties in exchanging protectively marked information. This has now been resolved however and I anticipate that we should be able to reach a meaningful conclusion during Step 4.

73 To conclude, I am broadly satisfied with the claims and arguments as laid down within the current PCSR, however the design of the inner containment requires further considerable effort to provide us with a suitable level of comfort over the use of grouted in place tendons. In addition, the use of an approach with Eurocodes as the basis for design in conjunction with a Non-UK National Annexe is under detailed review and there are a number of technical areas which will require resolution ahead of our acceptance of this approach.

2.5.6 Additional areas for Step 4 Civil Engineering and External Hazards Assessment

74 During the Step 4 assessment, there has not been any substantial change to the areas of assessment. Rather, it has been a refinement of the assessment to identify those areas where a deep slice sampling of the design approach has been undertaken. In addition, a greater integration of assessment topics has led to a more focussed support into other disciplines.

2.5.7 Use of Technical Support Contractors

75 Technical Support Contractors (TSCs) have been engaged to assist with the Civil Engineering and External Hazards assessment work thus far and will continue during Step 4. Whilst the TSCs have undertaken detailed technical reviews, this has been under close direction and supervision by ND and the regulatory judgment on the adequacy or otherwise of the UK EPR Civil Engineering and External Hazards has been made exclusively by ND.

76 A number of contracts have been let to TSC's with specialist knowledge in particular areas. These have fallen into the following broad categories.

- ABS Consulting have provided support in the areas of external hazards and design process reviews.
- ARUP have provided support on design code development and application and code validation and application.
- Atkins have provided support on a number of topics, most notably on geotechnical considerations and on reliability of design codes.

77 Gifford have provided support on pre-stressed concrete design and construction.

78 Sandia National Labs have provided support on containment analysis and design.

2.5.8 Cross-cutting Topics

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

- Severe Accidents (affecting containment design).
- Dropped Loads.
- Qualification.

- Management of Safety and Quality Assurance (MSQA).

2.5.9 Integration with other Assessment Topics

80 There are a number of technical areas which have a significant interaction with Civil Engineering and External Hazards, as identified below.

- Internal Hazards - Load Definition
- PSA - Fragility derivation and claims.
- Severe Accidents - Load Definition
- Mechanical Engineering - Interface with major plant items.
- MSQA - Design Audits and Independent third party review
- Radiation Protection - Shielding

81 There are also links with all other topic areas which are less significant, but nonetheless important in achieving a holistic view of the design. Where necessary, these are referred to in the relevant sections of this report.

3 EDF AND AREVA'S SAFETY CASE

82 The primary document which presents EDF and AREVA's safety case is the Pre-construction Safety Report (PCSR), Ref. EA1. The key elements of the PCSR in so far as they relate to the areas of Civil Engineering and External Hazards are presented below.

Table 2: Summary of Chapters of the PCSR Covering Civil Engineering and External Hazards

PCSR Chapter	Title	Contents Relevant to this Report
1	Introduction and General Description	Overview of Plant arrangement and introduction to building functions
2	Generic Site Envelope and Data	External Hazards considered in the design, rationale and magnitude
3.1	General safety Principles	General safety Principles
3.2	Classification Of Structures, Equipment and Systems	Safety classification of structures, rationale and application into design
3.3	Design of Category 1 Civil; Structures	Detailed description of design intent for civil structures
3.8	Codes and Standards used in the EPR Design	Overview of codes and standards
9.1	Fuel Handling and Storage	Overview of structures which house new and spent fuel
13.1	External Hazards Protection	Overview of how External Hazards are catered for in the design of EPR. Values used, and rationale for screening out combinations
15.2	PSA for Internal and External Hazards	PSA for Internal and External Hazards
15.6	Seismic Margin Assessment	Seismic Margin Assessment
20	Design aspects in relation to decommissioning	Decommissioning Strategy

83 In addition to the above, there are myriad 'Hypotheses' documents, codes and standards and internal technical guides which inform the design in more detail.

84 The key elements of the case as presented are as follows.

- The structures have been provided with an appropriate classification commensurate with the demands placed upon them.
- The design approach provides a level of robustness against the design loads commensurate with the requirements of the design classification.
- The structures are designed and capable of implementation such that the required through life performance can be assured.

85 Support to the deterministic case is provided via the PSA.

86 The civil structures requirements apply only to structures and are not applicable to systems or components.

87 Civil Structures have two main objectives:

- Protecting systems/components against hazards.
- Providing a barrier to the release of radioactivity.

88 Two requirement levels (C1, C2) are defined for Civil Structures as follows:

- Generally speaking, civil structures which house or support class 1 or 2 components or class 3 components which have a barrier role, are classified at Safety Class 1 and must meet C1 requirements. An exception is the turbine hall that is classified at Safety Class 2 and must meet C2 requirements.
- Civil structures which ensure a containment function are also classified as Safety Class 1 and must meet C1 requirements.
- Civil structures whose failure could impair the integrity of class 1 structures or those structures which house class 3 components, are classified as Safety Class 2 structures and must meet C2 requirements.

89 For C1 civil structures:

- The main structures (e.g. the Reactor Building internal containment, foundation raft…) must comply with the ETC-C design code and be seismically designed and constructed to SC1 requirements.

90 The other structures such as shielding protection inside the Reactor Building (see Sub-Chapter 3.2 - Table 5, which presents the list of these “other structures” in the Reactor Building and associated design requirement) must comply with dedicated design and construction rules and be seismically designed to SC2 as far as necessary. The anchorages of those “other structures” shall comply with ETC-C rules (see PCSR Sub-Chapter 3.8 for the ETC-C code).

91 For C2 civil structures:

- The structures must comply with dedicated design and construction rules and be seismically designed to SC2 as far as necessary.

The table below shows the mapping between civil structures safety classes and civil structures requirements levels.

Table 3: Summary of Civil Structure Safety Class

Civil Structure Safety Class	Civil Structure Requirements Level	Codes and Standards	Seismic Requirements
1	C1 (Main Structures)	ETC-C	SC1
	C1 (Other structures)	Dedicated Rules	SC2 as far as necessary (1)
2	C2	Dedicated Rules	SC2 as far as necessary (1)

(1) SC2 requirements apply to buildings/structures that protect or whose failure can have unacceptable impact on SSC with an SC1 requirement. In particular, if the collapse of a structure/building can directly or indirectly have unacceptable impact on SSC designed with an SC1 requirement (domino effect), this structure/building must be designed with an SC2 requirement. Unacceptable impact may result from the internal hazards subsequent to an earthquake (see Section 5.1.2).

92 The design requirements for civil structures are defined for the different load combinations considered in the design basis, including loads due to postulated earthquake (see PCSR Sub-chapter 3.3). The requirements cover the following aspects:

- Stability: behavioural requirements the purpose of which is to prevent the collapse of a civil structure.
- Local stability: behavioural requirements which are expressed in terms of static balance, mechanical resistance and rigidity.
- Integrity of equipment supports: behavioural requirements which describe the fact that the structural elements that support items of equipment must meet the requirements attributed to the equipment.
- Containment: the aim of the containment function is to limit the release of hazardous materials into the environment.
- Avoidance of interaction: the aim is to prevent impacts between adjacent components (including structures) during earthquakes. Interactions occur when the relative displacement of the components is greater than the separation distance between them.

Aircraft Protection

One of the other key loading scenarios relates to aircraft impact, and this is a key factor in the design of civil structures. Those structures which require physical protection have been identified.

93 The classification scheme is summarised in PCSR Chapter 3.2. Tables are provided which give greater detail for sub-structures and components for the Inner Containment. For other structures, this will be provided in future versions of the PCSR. The overall position however can be summarised as follows.

Table 4: Summary of the Classification of Civil Structures

Structure	Safety Class	Seismic Class	Aircraft Protected	Comments
Inner Containment	C1	SC1	Y (via APS)	
Safeguard Buildings	C1	SC1	Y (Via APS, Div 2,3 only)	
Nuclear Auxiliaries Building	C1	SC1	N	
Nuclear Auxiliaries Building Chimney	NC	SC2	N	
Fuel Building	C1	SC1	Y (via APS)	
APC Shell	C1	SC1	Y	
Waste Treatment Building	C1	SC1	N	

Structure	Safety Class	Seismic Class	Aircraft Protected	Comments
Diesel Building	C1	SC1	N	Separation of 2 buildings provides APC protection
CW Pumphouse	C1	SC1	Y (part)	
Turbine Hall	NC	SC2	N	
Tunnels/ Galleries	C1	SC1	Y	(Some buried beneath other structures)

94 A distinction is made between these buildings with regard to their design parameters however. For some of the structures above, the design is predominantly generic and therefore independent of the site in which they are installed; these include the reactor building, fuel building, safeguard buildings and the nuclear auxiliaries building. For some other structures, the design is site-specific, including the waste treatment building, the cooling water pumphouse and the tunnel network.

95 The second tranche of structures has not been considered in detail within GDA.

96 A more detailed description of the structures and their safety functions follows.

97 The bulk of the Nuclear Island structures are all founded on a common raft. This foundation raft is in the shape of a cruciform whose sides are about 100 m long. It forms the common base of the whole reactor building and the peripheral buildings, (the inner containment, fuel building and the four divisions of the safety auxiliary building). A corium recovery and cooling system inside the containment lower level is based in the reactor building raft which is located above the common raft.

98 The APC shell is designed to protect the Reactor Building, Fuel Building and Divisions 2 and 3 of the Safeguard Building against military and commercial aircraft crashes. It takes the physical shape of a thick wall which covers the roofs, and surrounds the outer walls of the Fuel Building and Divisions 2 and 3 of the Safeguard Building. The outer containment also provides the same protection at its dome and at the vertical upper section facing divisions of safeguard buildings 1 and 4. Additionally the vertical outer walls of the staircases for personnel access to the Nuclear Island buildings form columns which are part of the aircraft shell.

99 The Reactor Building is made up of a double-walled containment (and inner structures) located in the centre of the common base mat shared with the safeguard buildings and the fuel building which are located around the reactor building.

100 The inner containment is a pre-stressed concrete wall the inner surface of which is covered with a steel liner, which is embedded in the concrete at the foundation raft /Reactor Building internal structural boundary. A pre-stressing gallery is located below the raft for vertical pre-stressing tendons access. The inner containment wall is penetrated by electrical and mechanical penetrations, the largest of which is the equipment hatch through which heavy-duty reactor coolant system components are brought into the reactor building. The key role of the concrete structure of the inner containment is to withstand the over-pressures which may occur in accidents. The steel liner provides leak-tightness in these situations.

101 The outer containment wall is designed both to protect the inner containment from certain externally-generated hazards and to contain gas leakages from the inner containment, by means of the containment annulus ventilation system

102 The Safeguard Buildings are sub-divided into four divisions with their own access and containing each of the four safety trains. The trains comprise the mechanical and electrical systems and equipment needed to control fault situations that are taken into account in the reactor design together with the associated supporting systems, particularly the ventilation systems. The main control room is located in SAB 2.

103 A distinction is made between the two divisions of the safeguard buildings located between the reactor building and the turbine hall (divisions 2 and 3) and the other two located on each side of the reactor building (divisions 1 and 4) perpendicular to the axis formed by the reactor building and turbine hall. These two pairs of divisions are distinguished as follows:

- Divisions 2 and 3 are protected against certain externally-generated hazards by an aircraft impact resistant shell.
- Divisions 1 and 4, which are not designed against aircraft impact, contain the SIS rooms of trains 1 and 4 together with both trains of the corium cooling system (located in the CHRS rooms). The upper sections of these divisions support, on two different levels, the water and steam pipelines of the main secondary system and the associated isolation valves.

104 The Nuclear Auxiliary Building does not house Class 1 or Class 2 systems but does house auxiliary systems needed for reactor coolant system chemistry control, which may potentially be contaminated. Therefore, its structure performs the function of containment of radioactive materials that could potentially be released by failure of the systems and tanks which it contains. It has its own foundation raft adjacent to safeguard building 4 and the fuel building.

105 The effluent treatment building contains all the equipment necessary for the treatment of the contaminated fluids before their release to the environment or storage for transportation off-site. The design approach for the effluent treatment building is similar to that of the nuclear auxiliary building, since, as it contains radioactive products arising from the treatment of contaminated fluids, its structure must perform the function of retaining radioactive materials in case of failure of the systems and tanks which it contains. It is seated on its own foundation raft.

106 The Four diesel generators are installed in two buildings which are geographically separated to ensure redundancy in case of aircraft impact. Each of these two buildings contains two main emergency diesels together with one ultimate emergency diesel. The internal layout of these buildings is designed to avoid the risk of common mode failure of two diesel generators. Each of these buildings has its own foundation raft.

107 The pumphouse houses all the systems necessary for cooling both the nuclear and conventional plant. The pumphouse comprises a set of civil structures (concrete walls and structural steelwork) and equipment which provides coarse and fine filtration of the cooling water, and transfers it to the various pumped systems. The pumphouse installation comprises four divisions containing safety trains, which are separated by walls that protect the trains from common mode failure (especially flooding). The trains are supplied by diverse filtration systems. Divisions 1 and 4 of the structure are protected against commercial aircraft crashes. The pumphouse has a connected outfall structure whose role is to discharge plant cooling water to the sea (from both the nuclear and conventional islands) after it has performed its cooling duty, and to provide the fire

system water reserve. The outfall structure is seated on a foundation raft separated from that of the pumphouse. The pumphouse is a site specific design, it is excluded from GDA.

108 There are tunnels which run between the various buildings which contain all classes of systems. Their geographical location is designed to ensure that they meet criteria for protection against common mode failure with respect to externally generated hazards, particularly aircraft crash, earthquake and flooding.

4 GDA STEP 4 NUCLEAR DIRECTORATE ASSESSMENT FOR CIVIL ENGINEERING AND EXTERNAL HAZARDS

4.1 Assessment Strategy

109 The objective of the Step 3 and 4 assessment is to review the safety aspects of the proposed EPR designs as detailed in the PCSR. The primary guidance for the assessment is provided within the SAPs (Ref. ND5). Ref. ND5 was reviewed to produce a SAPs subset for this topic area. In considering the SAPs to be addressed (selection and sampling), the guidance contained in Refs ND6 and ND7 was followed, for example:

- I was selective in its confirmation of SAPs coverage in Step 3 and 4, e.g. through confirmation of credible claims and supporting arguments for the key SAPs identified by the lead assessor in each topic area.
- Judgement was used in selecting those SAPs for assessment at Step 3 and 4 and the level of detail to which the assessment was undertaken. The focus was on the systems leading to the largest risk reduction in addition to any systems employing novel or complex techniques.
- A mind map of the SAPs and their interrelationship can be seen in Figure 3 and a tabulated version in Table 2.

110 Assessment during Step 4 has tried to address the adequacy of the evidence supporting the arguments and claims identified in Steps 2 and 3. It is difficult to be absolute in the dividing line between arguments and evidence and in some areas the Step 4 assessment has clearly encompassed elements of what would normally be expected to be covered in Step 3.

111 Technical Support Contractor(s) (TSC) have been engaged to assist with the assessment work. Section 2.3.9 provides further details.

112 It is recognised that the designs being considered are international in dimension, and that they have been, or are being scrutinised by other nuclear regulators. Reviewing what overseas regulators have done and how HSE can make use of it has been undertaken in Step 4. Section 4.3 contains more information.

113 Finally, in summary, the key activities undertaken during Step 4 are as follows.

i) Assessment of responses to Step 3 observations.

ii) Set up and management of TSC support.

iii) Assessment of the arguments and evidence.

iv) Identification and management of relevant GDA related research.

v) Review of the results of other regulators' activities.

vi) Identification of GDA Issues and assessment findings.

114 In order to manage the tasks in a practical manner, the workscope has been broken down into a series of key areas, which are identified in the following paragraphs. The interpretation placed on the breakdown between argument and evidence, which is the key separator at a strategic level between Step 3 and 4 is also detailed in the following paragraphs.

4.1.1.1 Design Classification and Load Schedule

115 Within Step 3 the classification and load schedule were assessed at a principles level. In other words are they suitable for the design of nuclear safety structures. Key questions included:

- Does the classification scheme provide an appropriate link from safety requirements to design implementation?
- Has the load schedule been developed in a clear and consistent manner such that the safety requirements can be met?

116 Within Step 4 the application of the classification scheme and load schedule into the design was tested to confirm that they have been applied in an appropriate manner.

4.1.1.2 Codes

117 Within Step 3 the codes were assessed at a principles level. In other words are they capable of being used to design nuclear safety structures. Key questions include:

- Have the codes been developed and tested with sufficient rigour?
- Can the codes deliver the required levels of structural reliability?
- Do the codes deliver structures which are sufficiently robust?

118 Within Step 4 the application of the codes into the design has been tested to confirm that they have been applied in an appropriate manner.

4.1.1.3 Analysis

119 Within Step 3 the analysis tools (in so far as they had been identified) were assessed at a principles level. In other words are they capable of being used to analyse nuclear safety structures against the key demands placed upon them. Key questions included:

- Have the codes been developed and tested with sufficient rigour?
- Are the codes technically robust?
- Have the analysis codes been benchmarked sufficiently to give confidence in their predictive capability?

120 Within Step 4, it has become clearer as we have sampled in more depth that the range of software used was considerably larger, and the interfaces much more complex than we had envisaged. As a result, for some codes, they have been assessed at a principles level and an application level in Step 4.

4.1.1.4 Implementation

121 In order for structures to meet their design intent, they need to be buildable (to the appropriate level of quality), inspectable and maintainable. During Step 3, limited review of the buildability of the design was undertaken. It was been primarily restricted to a review of the operational feedback from the two current EPRs under construction at Flamanville 3 and Olkiluoto 3. However, a more detailed review of the ability of the pre-stressing elements of the containment to be constructed, particularly the grouting operations has been undertaken in Step 4. This is due to the specialised nature of the operations, and the limited options for post construction remediation of the containment were the design not to be implemented as intended.

4.2 Siting and External Hazards

122 The assessment topic of civil engineering and external hazards is a broad and complex one, and it is difficult to separate out individual aspects in a ready fashion. The following sub-sections however try to identify key themes in the assessment and link them to others in a coherent manner.

4.2.1 Generic Site Envelope

123 Fundamental to the idea of a generic assessment is the development of what is termed a generic site envelope. This defines a benchmark against which design activities can be undertaken, and ultimately what any proposed site characteristics will be measured against.

4.2.1.1 Scope

124 The first key Step in addressing the threats from external hazards is to identify those that are of relevance to the facility under consideration. This process is normally undertaken once a physical location for the facility has been established. However, for the GDA process, this is not the case. Hazards fall into one of the following categories:

- Hazards which will be present on all sites, and for which a design value has been estimated. This design value may be compared to the prevailing site conditions in the UK, to establish its reasonableness.
- Hazards which will be present on all sites, the magnitude of which cannot be determined until a site has been established, i.e. flooding, industrial hazards.
- Hazards which may be present on a site, but this cannot be established until a site has been selected.

125 The key Steps undertaken during the assessment are as follows:

- Review the process for Hazard identification and outcomes.
- Identify any hazards which have been screened out on the basis of either being:
  i) Non credible;
  ii) Low Frequency; or
  iii) No consequence.
- Identify those hazards which have been ruled out of specific consideration until a site has been identified.
- Review the above for conformance with SAPs.

4.2.1.2 Standards

126 The SAPs contain a specific section on ‘The Regulatory Assessment of Siting’, and includes 7 principles. This section of the SAPs as its title suggests is not geared towards the assessment of generic siting information; however there are some useful points which can be gleaned from it, principally key considerations over threats to nuclear safety which may be present on or near to a site. These are specifically:

- Metrology.

- Topography.
- Hydrology.
- Geology.
- Adjacent sites.

127 More useful guidance can be found in the section of the SAPs on External Hazards (EHA.1 to EHA.16). The Technical Assessment Guide T/AST/013 'External Hazards' (Ref. ND7) provides more detailed information on regulation of external hazards.

128 The International Atomic Energy Agency (IAEA) Documents NS-G-1.5, 1.6 and NS-R-3 provide additional guidance. (Refs OD3, 4 and 5).

4.2.1.3 Assessment – External Hazards Identification and Screening

129 Section 2 of the PCSR identifies the generic site that the UK EPR has been designed against. The hazards that have been taken into direct account are:

- Earthquake.
- High Wind.
- Tornado.
- Extreme Air Temperatures.
- Snow.
- Lightning.
- External Explosion.
- Malicious Activity.

130 The following site hazards have been recognised, but judged only capable of practical consideration once a site has been identified:

- Rainfall.
- Flooding.
- Biological Fouling.
- Infestation.

131 The following hazards have been dismissed as not worthy of further consideration at a generic level:

- Electro-Magnetic Interference (EMI).
- Ship Collision.
- Industrially Generated Missiles.
- Off Site Chemical Releases.
- External Fires (Brush fires etc).

132 Chapter 13 of the PCSR however does contain some further details on the above hazards including magnitudes to be adopted for the design which should then be confirmed for individual sites.

133 In my view this is an appropriate treatment in principle at the generic level. A more detailed investigation into the screening approach adopted for external Hazards has been undertaken by ABS Consulting (Ref. TSC4). This has highlighted the following three key points.

- A formal process for identification of external hazards is not evident, the approach being an historical one. The contents of the hazard listing has previously been agreed with the French and German national regulators.
- ETC-C would appear to contain a reasonable list of loads and load combinations, including those from external hazards.
- The PCSR recognises that not all external hazards can be defined until a specific site is identified. External flooding falls into this category. Seismic loading is addressed against a generic spectral shape, pending site-specific hazard data.

134 The lack of a clear process for identification and sentencing of hazards is surprising, however it has been decided to review the list of hazards against what would reasonably be expected as key considerations in the UK.

135 It is considered that the list of key external hazards which have been carried forward into the detailed design of the EPR are appropriate for the UK. In addition, those hazards which have been identified as only capable of detailed consideration once a site has been chosen are deemed to be appropriate for the UK.

136 It is considered however that the treatment of certain hazards which originate off site will require further consideration once a site has been chosen. In addition, the magnitude of all the hazards considered generically will need to be confirmed as appropriate once a site has been identified (Assessment Findings **AF-UKEPR-CE-01** and **AF-UKEPR-CE-02**).

137 The magnitude of the hazards used in the generic design has been reviewed at a high level and found to be broadly consistent with the Design Basis Events that we would expect for UK sites. However this will need to be confirmed during the site specific design (Assessment Finding **AF-UKEPR-CE-03**).

4.2.1.4 Assessment - Site Conditions

138 The UK EPR design has been undertaken against a variety of site conditions, as detailed in Section 13.1 of the PCSR. This defines 6 different sets of ground conditions against which the design has been assessed. The ‘soft site’ designated SA is considered to be slightly harder than some existing UK sites, especially those with large depths of estuarine deposits. The hard site envelopes the bulk of likely UK sites (see Figure 4). This issue will require much more detailed review at the site licensing stage (Assessment Finding **AF-UKEPR-CE-04**).

4.2.1.5 Summary

139 There has not been a clear and consistent process for the identification and screening of hazards, however the list of design basis events considered in the UK EPR design is considered reasonable.

140 The magnitude of the hazards used as design basis events are seen as reasonable for typical UK sites, however this will require much more detailed review at site licensing stage.

141 The range of soil conditions used in the design of the UK EPR is considered broadly representative of most UK sites, however this will require much more detailed review at site licensing stage

142 A series of assessment findings have arisen, as detailed below.

AF-UKEPR-CE-01: *The licensee shall examine the potential for EMI, Industrial hazards, transport threats, fire and release of chemical/toxic material from adjacent sites once a site has been chosen.*

AF-UKEPR-CE-02: *The licensee shall derive hazard magnitudes on a site specific basis for those hazards screened out as only capable of evaluation on a site specific basis, including rainfall, flooding, biological fouling and infestation.*

AF-UKEPR-CE-03: *The licensee shall confirm that the magnitude of all external hazards considered generically envelope those for the particular site under consideration.*

AF-UKEPR-CE-04: *The licensee shall confirm that for any structure designed using generic site data that that data is enveloped for the particular site under consideration.*

All of the above Assessment Findings (**AF-UKEPR-CE-01** to **AF-UKEPR-CE-04**) should be completed ahead of the placement of first structural concrete. This is to ensure that there are no options foreclosed for mitigation of the hazards which threaten the site as a result of concrete placed.

4.3 Civil Engineering

143 The following sections discuss individual; technical areas or individual structural elements of the UK EPR. There is clearly some degree of overlap between these areas.

4.3.1 Design Classification

4.3.1.1 Scope

144 Design classification is a major consideration for the whole of the EPR design; however it is rather simpler for the civil structures, as there are a limited number of classifications within the design.

145 The scope of this task covers all buildings which are being considered as part of the GDA (see Sections 4.2.2 and 4.2.3).

146 One other aspect which has also been considered is the classification of the external hazards which individual systems have been qualified against.

4.3.1.2 Standards

147 The key SAPs which are applicable to this area are as follows.

Engineering principles: safety classification and standards	**Safety categorisation**	**ECS.1**
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.		

Engineering principles: safety classification and standards	Safety classification of structures, systems and components	ECS.2
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.		

4.3.1.3 Assessment

148 Sub-chapter 3.2 of the PCSR provides a detailed description of the principles of the classification system adopted, with the requirements being graduated according to the importance of the safety duty being performed.

149 Section 3 of this report gives an overview of the classification of structures. This is primarily related to the function of the structure in terms of containing radioactive material, or in protecting the systems within a building from release of radioactive material and against postulated accident/ design basis events.

150 Clarification of the definition of the seismic classification system for civil engineering structures has been sought through GDA TQ-EPR-058. The response has provided some further clarity, however there was some uncertainty over the logic for the actual seismic design levels adopted for the various structures at Flamanville. It has now been made clear that the turbine hall has been designed for the local site conditions, the main Nuclear Island for the EPR envelope conditions and the remaining non-safety-related structures against local French building and safety authority requirements.

151 The most recent version of the PCSR provides clarity on the classification scheme for the main structures and for the substructures within the Inner containment (Tables 4 and 5 of Section 3.2 of the PCSR) Future versions of the PCSR will extend the detailed classification to other structures.

152 Having defined the system, the PCSR includes tables that list the chosen classification for:

- Main mechanical systems (Sub-chapter 3.2, Table 1).
- Main electrical systems (Sub-chapter 3.2, Table 2).
- Main fuel handling and storage systems (Sub-chapter 3.2, Table 5).
- I&C systems and equipment (Sub-chapter 3.2, Table 3).
- Civil engineering structures (Sub-chapter 3.2, Tables 4 and 5).

153 In addition, Section 13.1.1 - Table 1 of the PCSR provides a synopsis of the external hazards against which the individual systems have been designed. In general, most safety-related systems are designed against all external hazards. A review of the classification has not given any cause for concern.

4.3.1.4 Summary

154 The classification system is simple to understand, and for the major civil structures the rationale is clear. Future revisions of the PCSR will provide further details on the detailed classification for sub elements of the structures.

4.3.2 Load Schedule

155 The design of any civil structure requires a clear definition of the loads it should be capable of withstanding. This is typically defined as the load schedule. It typically follows from the design classification and safety case claims requirements, and is usually codified in the design standard. It is clearly linked to the nature of the site upon which the structure will be located (see Section 4.2.1). There is therefore some overlap with the findings in Section 4.3.3.

4.3.2.1 Scope

156 The following key Steps in assessing the load schedule have been identified:

- Identify load schedule(s) for key safety critical structures.
- Identify if all significant combinations have been considered appropriate though application of the SAPs.
- Review link back to functional performance and ensure consistency.
- Ensure that appropriate processes for development of loading parameters has developed and followed.
- Review application of loads in design.

4.3.2.2 Standards

157 The key SAPs which are applicable to this area are as follows:

Engineering principles: external and internal hazards	**Identification**	**EHA.1**
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.		

Engineering principles: external and internal hazards	**Design basis events**	**EHA.3**
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.		

Engineering principles: civil engineering: design	**Loadings**	**ECE.6**
For safety-related structures, load development and a schedule of load combinations within the design basis together with their frequency should be used as the basis for the design against operating, testing and fault conditions.		

Engineering principles: external and internal hazards	**Operating conditions**	**EHA.5**
Hazard design basis faults should be assumed to occur simultaneously with the most adverse normal facility operating condition.		

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

4.3.2.3 Overview of Load Schedule Development

158 In support of this task ABS consulting have been undertaking a review of the PCSR and the supporting design documentation to extract the relevant information, and have provided Ref. TSC40.

159 The starting point for the assessment was the identification, screening and selection of hazards for use in the detailed design of the EPR.

160 The global approach for accounting for external hazards is presented in PCSR Sub-chapter 3.1 Section 1.2.3.5.1. EDF and AREVA claims that hazards considered in the UK EPR design have been identified through several Steps, the main ones being:

- Consideration of experience feedback from current plants in France and Germany.
- Comparison with external hazards defined in the European Utility Requirements – Chapter 2.1 (Ref. OD6).
- French and international operational experience feedback during the development of the EPR design.
- Consideration of possible combinations of hazards.
- Consideration of hazards which may be generated by "malevolent acts."

161 As has already been said there is no evidence of a definitive list of all external hazards identified prior to any form of screening. Equally, there is no evidence of any formal screening process and hence the basis of the screening process.

162 Sub-chapter 15.2 of the PCSR (PSA) presents the results of an initial study to analyse the risk of core damage associated with internal and external hazards for the UK EPR. In the preface, EDF and AREVA claims that the set of hazards analysed correspond to those presented in Sub-chapter 3.1 Section 1.2.3.5. An external hazard is 'screened in' to the PSA if:

> "The consequences *of the external hazard could be important (to the plant structures, plant cooling systems, etc) and the hazard analysis frequency is not bounded by an internal event analysis already performed in the level 1 PSA.*
>
> *A detailed analysis is necessary to evaluate the frequency of core damage due to the external hazard."*

An external hazard is 'screened out' of the PSA if:

> *"There is no impact expected on the plant safety.*
>
> *The levels of defence are judged sufficiently efficient to give a low frequency of core damage.*
>
> *The frequency of the external hazard is low (10^{-5}/y)".*

163 The PCSR recognises that not all external hazards can be defined until a specific site is identified. External flooding falls into this category. There are some general principles

identified for flood hazard protection however. Seismic loading is addressed against a generic spectral shape, pending site-specific hazard data

164 Those external hazards which have been selected for detailed consideration in the design are listed below:

Variable Actions:

- **Variations in Temperature**
  - Air.
  - Water (normal and exceptional).
- Wind.
- Snow.
- Earthquake (Serviceability).

Accidental Actions:

- Earthquake (“Design Basis Earthquake”).
- Aircraft Impact.
- Vibration effects from each of the above.
- External Explosion (generic blast wave).

4.3.2.4 Design Application

165 Sub-chapter 3.3 of the PCSR covers the design of Category 1 civil structures. As previously noted, Sub-chapter 3.8 of the PCSR specifies that ETC-C applies. Hence Sub-chapter 3.3 can be viewed as an overview of ETC-C.

166 Sub-chapter 3.3 Section 1.3.4 states that:

> *“The external hazards considered for the design of the civil structures are:*
>
> - *Earthquakes: these are sub-divided into two different categories, namely the design earthquake and the inspection earthquake.*
> - *Accidental aircraft crash: accidental load cases are taken into account with load time functions, Note, the general aviation load cases are bounded by the military aviation load case and malicious aircraft crash is addressed in the UK EPR design.*
> - *External explosions.*
> - *Rising groundwater.*
> - *External flooding.*
> - *Exceptional meteorological conditions (temperature, snow, wind, missiles induced by tornados, etc).*
>
> *It is noted that lightning strike and electromagnetic interference are taken into account in the design of the civil structures via construction provisions.”*

167 The list of external hazards above is considered reasonable as a basis for design. The necessary combinations, treatment of consequential effects and beyond design basis considerations is given a more detailed consideration below.

168 The magnitude of the hazards used in the design has not been related to a return frequency in UK terms. A brief review has concluded that the magnitude of the hazards is not inconsistent with those anticipated for UK sites.

4.3.2.4.1 Load Combinations

169 The ETC-C provides guidance on load combinations for design, including loads from internal hazards and normal operational and construction loads. This includes some guidance on coincident hazards such as wind and snow.

170 The design incorporates what is termed an 'Event based approach' Report ENSN040070 (Ref. EA2) provides further details.

171 Section 2 of Ref. EA2 states that:

> *"The EPR design strategy incorporates event-based issues using event-based approaches identified as such (earthquake event approach) and/or rules of combination between internal or external hazards and single initiating events (mainly loss of off-site power and earthquake, external flooding or weather conditions)."*

172 In addition to this Sub-chapter 3.1 Section 1.2.3.5.3 states that:

> *"Overall protection from external hazards is ensured by defining the load combinations to be applied to plant, systems and structures which may be affected. For certain external hazards, the "load combination" approach may be supplemented by an event approach."*

173 Sub-chapter 3.1 Section 1.2.3.5.5 states that:

> *"For the EPR, different potential combinations of hazards are analysed, based on evaluation of operating feedback. The analysis takes into account:*
>
> - *Combination of physical phenomena inherent in the hazard itself.*
> - *Combination of the hazard in question with potentially dependent events or internal or external hazards.*
> - *Combination of the hazard with independent internal or external initial conditions."*

174 This approach is re-stated in Sub-chapter 13.1 Section 1.3:

> *"The combined events considered include the following scenarios:*
>
> - *Combination of physical phenomena inherent in the hazard.*
> - *Combinations of the hazard considered with potentially dependent internal or external events or hazards.*
> - *Combinations of the hazard and independent internal or external initial conditions."*

We have questioned EDF and AREVA at length on the practical application of these rules, and they continually refer out to Ref. EA10.

175 This document tabulates event combinations, which include at least one external hazard, to be taken into account during the analysis. These load combinations are intended to cover both 'coincident' and 'consequential' hazards.

4.3.2.4.2 Consequential Hazards

176 TQ-EPR-056 tried to address the subject of 'consequential hazards'. In the response EDF and AREVA state:

> *"In this section, three categories of consequential hazards are identified in the design of the UK EPR, as listed below*
>
> - *Combination of physical phenomena inherent in the hazard or PCC/RCC itself.*
> - *Combination of the hazard considered and potentially dependent internal or external hazards or faults.*
> - *Combinations of independent hazards and/or internal events."*

Only the second category can be classified as forming a 'consequential hazard'. This is borne out in the reply to TQ-EPR-114. It states that:

> *"… includes three kinds of hazards/events synchronous occurrences (also considered in PCSR subchapter 13.1 section 1.3): (1) inherent physical phenomena, (2) potentially dependent events and (3) independent internal or external conditions. Consequential hazards as defined in the query belongs to the second kind."*

The reply to TQ- EPR-114 states that:

> *"LOOP (Loss of off-site Power) is a consequential PCC event postulated for certain global external hazards like earthquake and wind. It is superposed in the analysis of consequential hazards."*

177 A list of consequential hazards addressed in the EPR design for each external hazard is given in the Table 2 embedded in the reply to TQ-EPR-114 shown as Table 5 overleaf.

178 There is an acknowledgement in the reply to TQ-EPR-114 that "The principles for addressing consequential hazards and the detailed exposition of those principles into design guidance as requested in this query is not presented in the current issue of the PCSR." The most recent version of the PCSR (March 2011), Ref. EA253, does however cover this aspect.

179 The reply to TQ-EPR-114 also states that:

> *"When the identified consequential hazards may lead to unacceptable consequences, design measures are taken so that consequential hazards can be ruled out ("decoupling"), or effects of consequential hazards can be mitigated to an acceptable level i.e. it is ensured that the general rules for internal hazards as presented in Sub-chapter 13.2 Sub section 1.2.1 are met at all times"*

180 Logic also suggests that the combinations not captured in the above table are 'coincident' hazards and fall within EDF and AREVA categories (1) or (3); namely:

- Combination of physical phenomena inherent in the hazard itself.
- Combination of the hazard with independent internal or external initial conditions.

Table 5: Summary of the Consequential Hazards Designed for in the EPR

Initiating External Hazard	Consequential Hazard
Earthquake	Fire

(+LOOP)	
(+LOOP)	All remaining Internal Hazards
	External Flooding
	External Explosion
Industrial Risk (Expolsion)	Internal Flooding
	External Flooding
Wind (+LOOP)	Wind Generated
Lightning	Fire, internal explosion

4.3.2.4.3 Hazards in Design

181 The PCSR document and the ETC-C are insufficient to allow a designer to proceed with the detailed design of the civil structures. EDF and AREVA have developed a series of 'Hypothesis documents' which are essentially detailed design guides which extract the necessary details from the PCSR and elsewhere to allow the designer to proceed.

182 Figure 3 shows the 4 levels of documents which have been identified through the assessment. As part of the review of these hypothesis documents several key points have become apparent.

183 Screening of hazards at a detailed level is undertaken and detailed at the lowest level of hypothesis document. An example is the lack of need to consider wind loading for the inner containment structure as it is shielded by the aircraft shell, other than during construction. This can be seen to be a sensible approach, however it is surprising that such a principle were not established at a higher level within the documentation. It leaves open the question of whether decisions taken at a lower level in the process are consistent with the principles laid down in the PCSR. This is perhaps a reflection of the timing of the production of the various documents which detail the EPR design approach.

184 As part of the Step 4 review a more detailed sampling of the Level 2, 3 and 4 hypothesis documents has been undertaken as well as a review of the process by which the design intent is preserved through the trail of documentation and the process by which EDF and AREVA ensure that the design house approaches are consistent with the overall philosophy.

185 Sub-chapter 3.3 Section 1.3.5 also talks about the inclusion of margins in the design of the EPR civil structures.

186 Sub-chapter 3.3 Section 1.3.5 states that:

> *"External events, where the design of the structures must make provision for loadings, whether they are due to natural phenomena (i.e. earthquakes or climate change) or human induced events (e.g. explosions or aircraft crash)".*

187 The design also takes into account a double-ended guillotine break of the reactor coolant pressure boundary (i.e. 2A LOCA) and the combined loading due to a simultaneous loss of coolant accident (i.e. reactor coolant system pressurizer surge line break LOCA) with the design earthquake. The purpose of designing against this load combination is to ensure that substantial margins are present in the design of the inner containment lower section." This is discussed in more detail under the containment section of this report.

188 This explains the inclusion of the SLLOCA+DE event in ETC-C AFCEN Table 1.3.3.2. It is also noted that a Seismic Margins Assessment (SMA) is reported in Section 15.6 of the

PCSR, in the absence of the site-specific seismic hazard to use in the Probabilistic Safety Analysis (PSA). The SMA targets a High Confidence Low Probability of Failure (HCLPF) value of 1.6 times the design basis earthquake. However, it is not yet apparent how loadings beyond the design basis for other hazards have been considered in general. This is reviewed in more detail later in this report (see Section 4.3.9).

189 The treatment of drop loads within the design is not considered adequate. Various hypothesis notes state that this will be considered, however there is no evidence to support this. As a result, a GDA Issue **GI-UKEPR-IH-01** has been raised under the Internal Hazards topic area (see Ref. ND27). This is primarily concerned with the identification of drop load scenarios which require consideration in the design of structures, however one of the actions relates to the derivation and application of structural design methodology. This has potential impacts on the design of aspects of the civil structures.

190 The assessment of the EPR Internal Missile Methodology, the specific issue of the break preclusion claim made against RCC-M classified vessels, pumps, tanks and valves was identified as requiring further detailed assessment from both an Internal Hazards and Structural Integrity perspective and has been raised as GDA Issue **GI-UKEPR-IH-04** (see Ref. ND27). This has potential impacts on the design of aspects of the civil structures.

191 It is therefore an assessment finding that the licensee shall take account of any implications of the outcomes of the Internal Hazards GDA Issues which could affect the design of civil structures (Assessment Finding **AF-UKEPR-CE-05**). This finding should be completed ahead of the placement of first structural concrete. This is to ensure that there are no options foreclosed for mitigation of the hazards which threaten the site as a result of concrete placed..

4.3.2.5 Summary

192 A formal process for identification and screening of external hazards is not evident, the approach being an historical one, however, the list of external hazards considered in the EPR design appears to be reasonable.

193 ETC-C would appear to contain a reasonable list of loads and load combinations, including those from external hazards.

194 The PCSR recognises that not all external hazards can be defined until a specific site is identified. External flooding falls into this category. Seismic loading is addressed against a generic spectral shape, pending site-specific hazard data.

195 Load combinations are to be found in ETC-C, supplemented by event-based combinations.

196 ETC-C includes coincident hazard load combinations, such as from wind and snow.

197 The “Event Based Approach” includes coincident hazards. This covers both physical phenomena inherent in the hazard (such as external flooding coincident with rainfall and high water table), and combinations of the hazard with independent internal or external initial conditions (such as the choice of temperature-dependent material properties for the earthquake loading condition).

198 The “Event Based Approach” includes consequential hazards, such as fire following earthquake.

199 The PCSR states that the design of the structures must make provisions for loading beyond the design basis, although in general it is not evident how this is accomplished. It

is noted that a Seismic Margins Assessment (SMA) has been performed with a target of the HCLPF value being not less than 1.6 times the EUR design basis ground motion spectrum.

200 The approach to identification and screening and production of loads for consideration in the design does not follow a conventional approach, the outcomes in terms of requirements on designers are considered to be adequate.

201 In summary, whilst the approach is somewhat different than would be traditionally expected in the UK, it is considered that the outcomes are satisfactory and aligned with our expectations.

202 The following Assessment findings have been identified:

> ***AF-UKEPR-CE-05:*** *The licensee shall take account of any implications of the outcomes of the Internal Hazards GDA Issues which could affect the design of civil structures. This finding should be completed ahead of the placement of first structural concrete. This is to ensure that there are no options foreclosed for mitigation of the hazards which threaten the site as a result of concrete placed.*

4.3.3 Design Codes

203 Each of the Civil Structures in the reference design has been designed using the EPR Technical Code-Civil (ETC-C), Ref. EA3. This is an EDF and AREVA specific code, developed for the EPR project. The ETC-C is essentially a signposting document which directs the designer to assorted Eurocodes, French standards and other guidance. It also contains specific modification to the normal Eurocode approach in some areas, and specific design guidance in others.

204 For the UK, a revised version of ETC-C has been presented, ETC-C AFCEN 2010 (Ref. EA73). For the purposes of GDA, while the ETC-C Rev B was included in the assessment process, it is the later ETC-C AFCEN document that is being assessed as being the GDA design code, along with its UK companion document (Ref. EA74). It should be noted that some of the deep slice evaluations of structures have examined designs undertaken to ETC-C Rev B. The shift in proposed design code is reflected in the assessments and subsequent assessment findings however. The following paragraphs describe the process which has been followed to arrive at the current position.

4.3.3.1 Scope

205 The following key Steps have been identified:

- Identify the principles by which codes have been selected for application against functional requirements and review against SAPs.
- Review the codes used for the following:
  - o Application to Nuclear Structures.
  - o Development History.
  - o Currency (i.e. are they up to date and supported).
  - o Derogations from standard application.
  - o Previous regulatory interactions (UK and elsewhere).

- o Code Interfaces.
- o Reliability targets.
- o Degradation/ lifetime allowances.
- Review the application into selected structures:
  - o Containment.
  - o Reactor Buildings.
  - o Essential Plant Buildings.
  - o Diesel Buildings.
- Review code application in light of above.

4.3.3.2 Standards

206 The key SAPs which are applicable to this area are as follows.

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.		

207 Further guidance on the development of codes and standards is also contained within British Standard (BS) 0 'Standard for standards' (Ref. OD11). Whilst this is specifically written for the development of UK standards it nonetheless presents principles for the production of documents which are considered to be national standards.

4.3.3.3 Overview of ETC-C

208 The ETC-C has its origins in the earlier EDF Code RCC-G. A brief history is included below.

- First RCC-G EDF edition: December 1980 (for 900 MWe NPPs)
  - French Safety Authority Approval : Basic Safety Rule RFS V.2.b July 1981
- Following EDF editions released:
  - 2nd Edition: January 1986
  - 3rd Edition: 1985
  - January 1985 Technical Specifications for 1300 MWe NPPs.
  - Safety Authority Approval: RFS V.2.h June 1986
  - Modification sheets : RFS V.2.h –rev. 1 Oct. 1988
  - AFCEN Edition
  - July 1988 (French and English languages)
  - 1992: EDF document

Evolution of ETC-C

- Initial Rules: 1995 (NPI)
- Later rules.
  - 1999 Design rules for Containment with partial composite liner
- GPR/RSK recommendations.
  - 2001: EPR Containment with steel liner
- Criteria for containment (2003)
- Safety Authority Approval: July 2004
  - 2004-2006: ETC-C with design, construction and tests Parts 1, 2 and 3
  - April 2006 last version
- AFCEN 2010 version Issued (Ref. EA75).

209 The ETC-C is comprised of 3 basic parts.

Part 1: Design

- Actions and combinations of actions.
- Concrete structures (criteria from EC2 + complements)
- Metal parts contributing to leak-tightness (containment liner and penetrations, pool liners)
- Steelwork and plate anchorages.
- Annexes: seismic analysis, shrinkage and creep, simplified methods for impact (military aircraft) and perforation calculations.

Part 2: Construction

- Materials: soil, concrete, formwork, rebars, pre-stressing, precast.
- Penetrations, liners for containment and pools, steelwork.
- Tolerances for procurement, construction.

Part 3: Instrumentation (monitoring) and tests

- Leak tightness tests.
- Instrumentation and mechanical resistance tests.

210 ETC-C Content.

The following aspects are included in ETC-C:

- Design and Construction Rules and Criteria for C1 structures (general rules and rules for containment).
- Metal parts embedded in containment (Including liner, penetrations sleeves and Equipment Hatch).
- Methods related to containment tests.

The following aspects are excluded from ETC-C

- Site data (earthquake, temperature, wind, etc.)
- Special Project requirements.
- Specific building actions and rules.
- Components covered by specific Technical Specifications (TS) such as:
    - Airlocks/Electrical penetrations.
    - Plates.
    - Metalwork.
    - Paintings.

- Relationship with constructors: procedures, checks and controls.
- Details of monitoring.

Safety requirements are detailed in ETC-C: Appendix 1.G (informative).

4.3.3.4 Assessment

211 The assessment of ETC-C has taken place through Step 3 and 4 of GDA and has been the subject of a large number of TQs, and technical meetings. This has resulted in the final assessment for GDA considering the AFCEN 2010 version of ETC-C along with the UK companion document (Refs EA73 and EA74).

212 The ETC-C relies heavily on Eurocodes for detailed design rules which it supplements with additional guidance. It is useful to have an overview of the role of Eurocodes and the framework in which they operate.

213 The 58 parts of the Eurocodes are published under 10 area headings. The first two areas – basis and actions – are common to all designs, six are material-specific and the other two cover geotechnical and seismic aspects. Those of relevance to the EPR design are as follows (Refs OD15 to OD21).

EN1990 Eurocode 0: Basis of structural design

EN1991 Eurocode 1: Actions on structures

EN1992 Eurocode 2: Design of concrete structures

EN1993 Eurocode 3: Design of steel structures

EN1994 Eurocode 4: Design of composite steel and concrete structures

EN1995 Eurocode 5: Design of timber structures

EN1996 Eurocode 6: Design of masonry structures

EN1997 Eurocode 7: Geotechnical design

EN1998 Eurocode 8: Design of structures for earthquake resistance

The ETC-C does not make reference to Eurocodes 5,6 or 9.

214 Each published part is referenced by the standards body identifier (e.g. BS in the UK) followed by the EN code prefix, part number and year published (e.g. BS EN 1991-2: 2003). This is then followed by a full title (e.g. Eurocode 1: Actions on structures – Part 2: Traffic loads on bridges).

215 The national standard implementing each part comprises the full, unaltered text of the Eurocode and its annexes as published by the European Committee of Standardization (CEN). This is preceded by a national title page and a national foreword and is followed by a national annex.

216 Safety remains a national and not a European responsibility; hence the safety factors given in the Eurocodes are recommended values and may be altered by the national annex. Possible differences in geographical or climatic conditions (e.g. wind or snow maps) or in ways of life, as well as different levels of protection that may prevail at national, regional or local level, are taken into account by choices left open about values, classes, or alternative methods called ‘nationally determined parameters’. They allow EU member states to choose the level of safety, including aspects of durability and economy applicable to works in their territory, through their national annex.

217 A Eurocode part is not ready for use in a country until its national annex is published, which typically follows within a year of the part's publication. Published national annexes are referenced as 'NA to' followed by the part reference.

218 It should be noted that the Flamanville 3 reference design has been undertaken using the French National Annexes to the Eurocodes.

219 Within the UK, HSE is mandated to apply building regulations to Nuclear Licensed sites. The Building Regulations 2000 (as amended) set out the kinds of work that are exempt from the Regulations:

> *"Any building (other than a building containing a dwelling or a building used for office or canteen accommodation) erected on a site in respect of which a licence under the Nuclear Installations Act 1965 is for the time being in force"*

220 The requirements within Part A of the building regulations (2004) (Ref. OD23) can be met by the use of what are termed 'approved documents'. These are listed in the back of the building regulations, and are currently listed as extant British standards. It is not mandatory to use them, however they have a 'deemed to satisfy' status. It has been recognised however that these will be withdrawn in the near future to be replaced by Eurocodes and it is further stated that there will be periodic updates to the building regulations to reflect this:

> *In January 2010, the department of communities and local government wrote to all Building Control Bodies (BCB's) (Ref. OD22) advising the following "When assessing compliance with the Building Regulations, BCBs should continue to consider the appropriate use of relevant standards on a case by case basis. This may include the use of the new BS ENs, which formally become the new national standards in April 2010 reflecting the changes made by the standards organisations"*

> *Interpreting the current part A document and the letter addendum, it is clear that the use of Eurocodes with UK national annexes is an automatic satisfaction of the requirements. However it should be noted that it is not mandatory. In other words other standards and codes may be used. Approved document A states that "Thus there is no obligation to adopt any particular solution contained in an Approved document if you prefer to meet the relevant requirement in some other way."*

221 I have interpreted the above as a requirement to ensure that the design approach used provides at least as robust a solution as if the approved documents were used. In order to satisfy this, I have undertaken a considerable effort in examining and benchmarking the codes used. This is discussed in more detail in the following.

222 One specific requirement of the building regulations is around the subject of robustness and disproportionate collapse for which there are specific expectations. TQ-EPR-857 was raised, specifically asking for a demonstration that the robustness and disproportionate collapse requirements were met. The response asserts that the structured approach to internal and external hazards results in an approach which is more structured and rigorous than that required for Class 3 structures in Part A of the building regulations. I agree that this is an acceptable argument, however I still consider that specific consideration to robustness should be given in the design of the structures. This is reflected in the GDA Issue on the hypothesis notes for the Class 1 structures where a specific requirement to include this consideration is made. In addition, it is an assessment finding that the licensee shall undertake a review of all Class 1 and 2 structures to ensure that the minimum tieing requirements for Class 2 structures within the building regulations are met.

4.3.3.5 ETC-C Revision B

223 Within Step 3 of GDA, I undertook a systematic review of the development path of ETC-C rev B as well as examining the individual clauses and requirements, with particular focus on Part 1 of the code.

224 The key findings in Step 3 were that ETC-C is an EPR specific design code: it is not applicable for general construction. It is not equivalent to a standard design code and could not be applied by a designer unfamiliar with the design of this type of NPP. This is evidenced by the following.

- The background development to the code is not stated.
- The objectives laid out for the code are not made clear.
- There is a lack of a clear statement on the target reliability to be achieved through use of the rules.
- There is no evidence of benchmarking against other codes.

On a more practical level, there are a number of issues which affect the manner in which the code is used:

- There is a lack of clarity over which Eurocode rules may be used with and without modification.
- There are references to superseded design standards (e.g. Eurocodes) and Euronorms throughout.
- ETC-C does not provide prescriptive rules for all situations. As a result, there could be seen to be a lack of control over the use of alternative design methods and the means of ensuring that alternatives achieve the required level of reliability. This reinforces the requirement for additional documentation to be available to the designer, the nature of which is not always clear.
- The compatibility between the design rules and the workmanship rules needs further investigation as it appears that they may not be fully compatible at present.

At a lower level still, the detailed review of the code has identified that there are a number of issues such as:

- Errors in equation formulations.
- Errors in referencing.
- Potential deviations from the UK national annexe approach

225 These findings were supported by work undertaken by TSCs, Refs TSC7 andTSC11 and TSC12.

226 As a result of these findings, I wrote to EDF and AREVA (Ref. ND13) identifying these issues. I received a reply (Ref. EA71). The key points raised in this response were that:

- The ETC-C needs to be read with the particular hypothesis notes for the building under examination. Hypothesis notes are typically prepared at three levels, the highest level by EDF (CNEN), the second level by Sofinel, and the third and most detailed level by the individual design teams for the building in question.
- EDF and AREVA accepted that that the "*definition of design rules using multiple documents is not fully satisfactory. We are therefore proposing to produce a user*

guide to Part 1 of ETC-C, which will fully define all the design rules and the links between ETC-C and Eurocodes in a single document."

227 The one key issue which was raised as a Regulatory Observation (RO-UKEPR-037) is 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 you require through appropriate choice of not only design methods (partial factors), but also implementation control methods. The ETC-C is silent on this subject, and as a result, following discussions with EDF and AREVA an RO was raised (RO-UKEPR-037). The response to RO-UKEPR-037 is discussed in Section 4.3.3.7. Following the issue of RO-UKEPR-037, discussions were held and a set of actions agreed.

228 In addition to re-issuing RO-UKEPR-037, I held a workshop with a selection of technical support contractors to discuss other options for considering the reliability of the ETC-C code. This generated a new workstream to undertake some limited independent benchmarking of ETC-C. This benchmarking focussed on a small number of areas creep and shrinkage, durability, bending and axial load, shear without reinforcement, shear with reinforcement and crack widths. This was reported in Ref. TSC13.

229 The work reported in Ref. TSC13 identified the following key points:

- A high level comparison of detailing practice has indicated that the ETC-C is broadly consistent with both EN1992-1-1, as implemented by UK NA, and previous UK practice. However, clarification of the correct interpretation of some clauses in the ETC-C is required.
- Where the ETC-C follows the approach of EN1992-1-1, as implemented by UK National annex, it can be assumed the level of reliability will exceed SAPs' BSL requirements but may not achieve SAPs' BSO levels without further actions. These actions may include consideration of more onerous load cases or additional supervision and inspection during design and construction.
- In several areas, the ETC-C introduces alternative approaches to those in EN1992-1-1 used with the UK NA. These alternative approaches generally produced less conservative and therefore less reliable resistances. In some cases, the differences were so significant that calculating a reliability using the approach in EN1990 was not believed to be valid. All that could be concluded was that in many cases, probability of failure would increase by an order of magnitude when compared to EN1992-1-1.
- Insufficient research has been found to verify that the methods adopted by the ETC-C would nevertheless comply with SAPs' targets. Further work to justify these methods is required.
- In some areas it was not clear how the alternative rules proposed by the ETC-C (and the French NA,) would be applied. In particular, the calculation of creep and, hence, pre-stress losses, crack width limits for compatibility with liner design and durability requirements, and the minimum shear capacity of slabs require further consideration.
- There are a number of issues, related to the resistance models, which required further justification

- The appropriate value for a_{cc} to be used in axial load and flexure calculations is less conservative in the ETC-C (and the French National Annexe) than the UK National Annexe.
- The limit on concrete strength used in shear calculations and its reliance on specific testing of actual concrete mixes.
- Clarification of creep model/modular ratio to be used in design is required.
- Clarification of the application of increased minimum shear strengths for slabs with transverse shear distribution introduced by the French NA to EN1992-1-1 is required, Clarification of the application of increased minimum shear strengths for walls is required.
- The method of calculating crack widths requires justification
- The use of g_c = 1 for shear in accidental load cases Groups 2 and 3 is inconsistent,
- The Implications of changes to EN1992-1-1 on pre-stress losses should be investigated,
- There appears to be little justification for the introduction of g_c into the EN1992-1-1 minimum shear equations
- There appears to be no justification for using modified shear model when using shear reinforcement particularly for high strength concretes and deep sections.
- There is no justification for the alternative, to EN1992-1-1, concrete strength reduction factor for shear with reinforcement.

230 In mid-2010, EDF and AREVA issued the ETC-C UK application document (also known as the user guide), Ref. EA63. This was reviewed thoroughly, partly by TSC's (Ref. TSC14 and 27). As a result, I wrote to EDF and AREVA outlining my views on the user guide (Ref. ND19). This letter covered aspects raised in Refs TSC13, TSC14 and TSC27. The key items of concern raised were as follows.

- There is a confusion over precedence between the ETC-C and the user guide which should be clearly laid out in the front of the guide and in relevant sections as necessary.
- The term "user guide" is somewhat of a misnomer, as there is generally little guidance provided. It is typically a collection of corrections, errata and other changes to make the ETC-C comprehensible. It is understood that there was no intention to provide a detailed breakdown of the rationale behind the code choices, however there is little evidence of any input from designers experienced in the use of the ETC-C in the guide.
- The sections on the use of National Annexes are still somewhat confused. There should be absolute clarity in this area.
- In the longer term, a revision to the code is understood to be the final solution. Revisions to the layout of the code would benefit from some rationalisation to bring it more into line with the Eurocodes which support it.
- There are a number of extant and outstanding TQs which have not been addressed in the user guide.

231 During a series of technical meetings, the above comments were discussed, and EDF and AREVA declared that their strategy had changed and that a new version of ETC-C would be provided, via the AFCEN organisation, to be known as "ETC-C AFCEN 2010". This would be read in the UK in conjunction with a UK companion document, which would provide additional guidance and information for the UK only.

232 In order to provide further justification in the identified key areas above, EDF and AREVA have supplied a series of supporting reports (Refs EA54 to EA62). We have replied to these references under cover of Ref. ND20.

233 ETC-C AFCEN 2010 was published in December 2010.

234 There are a couple of areas of the ETC-C where clear resolution of the concerns were reached ahead of the publication of ETC-C AFCEN, which are detailed below.

235 Section 1.3.3.5.6 of the ETC-C Rev B and 1.3.3.3.9 of ETC-C AFCEN states that the protection of the structural components from fire is achieved by following the requirements of EN1992-1.2 for concrete and EN 1993-1.2 for steel. The concrete grade used for the containment structure is C60/75. For this grade, additional precautions are required under the fire limit state when the silica fume content exceeds 6%. The response to TQ-EPR-167 notes that the actual silica fume content will be approximately 10%. In TQ-EPR-286 it is noted that to satisfy EN1992-1.2 requirements specific fire tests will be carried out using the actual materials to be used in the construction. The licensee will need to undertake any necessary fire tests on reinforced concrete walls using the actual materials to be used in the construction in accordance with the requirements of EN1992-1.2 (Assessment Finding **AF-UKEPR-CE-06**).

236 Section 1.4.1 of the ETC-C states that the design working life of the structure should be 65 years. Eurocode 2 only gives recommendations for the requirements of structures with a design working life of 50 or 100 years. The operating life of the main structures may well be only 65 years, however they will undertake safety critical function for some time longer than this, up to 100 years for most structures and possibly longer if longer term storage of spent fuel is required on the site. TQ-EPR-627 and TQ-EPR-819 were submitted and responded to. The response is in tandem with those for decommissioning timescales and confirms that the design approach used ensures that the as designed lifetime is sufficient for the lifetime requirements. Assessment of the hypothesis notes also revealed a lack of consistency in the design lifetime being adopted. The development of UK specific hypothesis notes has been raised as a GDA Issue, and should include consideration of the design life.

4.3.3.6 ETC-C AFCEN 2010

237 For application in the UK, ETC-C AFCEN 2010 requires the UK companion document to be used alongside (Ref. EA74). We have reviewed the ETC-C AFCEN using the following approach.

238 Review the ETC-C AFCEN associated submissions (Refs EA54 to EA62) and outstanding TQs.

239 Review the revised ETC-C AFCEN 2010 against the following:

- How it addresses issues raised in the detailed review of the ETC-C User guide.
- How it addresses issues raised in TQs on the ETC-C which have been issued post April 2010, the effects of which was not captured in the previous user guide.
- How consistent is the User Guide with responses to TQs on the ETC-C.

- Have any new areas of concern been introduced.

240 Review the revised ETC-C AFCEN 2010 UK specific companion document.

4.3.3.6.1 Review of ETC-C AFCEN Supporting Reports

241 The review of the supporting reports (Refs EA54 to EA62) revealed a number of ongoing concerns (see Refs TSC45 to TSC54). These were transmitted to EDF and AREVA in Ref. ND 20. The key points from this letter are summarised below.

242 EN1992 introduces a term to modify the design strength of concrete. The term α_{cc} has a recommended value of 1.0 in part 1 (general) of EN1992, and a recommended value of 0.85 in part 2 (bridges). The UK National Annex adopts a value of 0.85, for bending and axial load, for both parts on EN1992, whilst the ETC-C takes a value of 1.0. Ref. EA54 proposes that the ETC-C adopts the UK approach except in certain accidental actions when a value of 1.0 is proposed. The document does not fully address the differences between EN1992 parts 1 and 2 and justify the choice for the EPR structures. Duration of loading is focused on as the main factor to be considered, with heat of hydration mentioned briefly, and the arguments on the effect of silica fume on concrete strength are hinted at the arguments are not fully developed.

243 Revision B of the ETC-C gives the loads and load combinations to be considered. Ref. EA55 does not split out the factors of safety on the load and the combination factors to be used. This is not consistent with the Eurocode approach but ETC-C AFCEN 2010 does use the Eurocode approach. . It would help transparency of the code if the two factors were presented separately and would have benefits for applying the ETC-C to particular areas where the loading frequency is different to the majority of the EPR. The development/justification of a full set of combination factors for the structures on the Nuclear Island and adoption of normal Eurocode load combinations would resolve these concerns.

244 Ref. EA56 sets out the justification for the use of a stress limit of $1.2fck/\gamma c$ for concrete under biaxial or triaxial stress states. There are no particular comments on the method presented however assumptions are made about the stresses in the other two directions to justify a maximum stress of $1.2fck/\gamma c$ in the third direction. These requirements are not embedded into the ETC-C.

245 Ref. EA57 sets out the basis for a proposed revision to the ETC-C for the calculation of shear resistance and shear reinforcement requirements. It starts with the background to equations used in EN1992-1-1 and then proposes an alternative approach. It appears that in the areas where link design is normally required the proposal is less conservative than that in EC2, using both recommended values and using the UK National Annexe values. EDF and AREVA have therefore been requested to adopt the equations in EC2 to define shear reinforcement requirements for UK EPR.

246 Ref. EA58 proposes increasing the steel material factor of safety in the ETC-C from 1.15 to 1.4 and from 1.0 to 1.25 for the normal Ultimate Limit State (ULS) and accidental ULS cases respectively. This is a significant increase but it is noted that the new factors are still lower than those derived from EN1992-4 when the likely steel properties are considered. The values are however in excess of those recommended in the CEB guide, and on this basis are deemed to be acceptable. The justification for not including normal reinforcement fasteners is not clear and the purpose of these fasteners and/or the justification for using lower material factors on them should be clarified.

247 Ref. EA58 sets out a rigorous basis for determining the contribution of horizontal tendons in corrugated ducts. It demonstrates that the ETC-C is conservative for group 2 and group 3 load combinations, although the approach for group 3 combinations could be stated more clearly in the ETC-C. The document does not justify the approach in the ETC-C for group 1 combinations however it is considered that the effects of the discrepancy are not significant. There is no discussion on the approach to vertical tendons, the approach when tendons are in smooth ducts, nor the approach close to singularities when stabilised cracking may not develop. Given the number of tendons in smooth ducts and the tendency for these to be adjacent to major penetrations, which are considered singular zones, the lack of specific rules in this area is an omission in the document and of the ETC-C.

248 The overall approach of the strain method in Ref. EA59 is acceptable, however the differential shrinkage should be calculated taking into account the other forms of shrinkage including early thermal and autogenous shrinkage effects. It is believed that the correct value for the k2 parameter should be used in the crack width equation. However the fact that the method uses the EN1992-2 method for calculating drying shrinkage, and for thinner elements this is more onerous than the EN1992-1-1 method, some conservatism is introduced that may offset the difference. The intended approach of using the strain method to check all elements and the Force method, in addition, to check leak-tightness needs to be clearly stated.

249 Crack widths are required to be controlled for various reasons including aesthetics, water resistance and compatibility with waterproofing membranes. For the EPR strain compatibility with the assorted pool and containment liners is a further consideration. In addition crack widths are controlled for durability purposes. The acceptable crack width criteria are confused in and contradictory in the document. Secondly, the approach for incorporating shrinkage into the calculation, and the use of k2 = 0.5 requires substantiation. This is correct for pure flexure but is non conservative when tension is present, given that shrinkage is thought to be a significant cause of cracking and this will produce predominantly tensile effects this is considered non-conservative. TQ-EPR-824 attempted to justify this approach, however not in a convincing manner. The acceptability of this non-conservative assumption requires consideration in the context of the overall shrinkage/cracking approach. Thirdly, the statements that the method in 1992-1-1 “can be completed and refined by different methods” appears to remove any restriction on the approach used and makes the document largely irrelevant.

250 The EDF and AREVA response to Ref. ND20 is in the form of a revised series of reports (Refs EA240 to EA248). I have been unable to undertake a detailed review of these documents within Step 4 of GDA. This forms part of GDA Issue **GI-UKEPR-CE-02**.

4.3.3.6.2 ETC-C AFCEN 2010- Assessment

251 AFCEN describes itself thus *“AFCEN was incorporated in October 1980. In accordance with its articles of association, its purpose is (i) to draw up rules for the design, manufacture, installation and commissioning of components intended for Nuclear Islands for electricity-generating purposes, (ii) to modify these rules in the light of knowledge acquired and of technical progress and new developments in technology and (iii) to publish works relating to these practices or amendments thereof. Finally, it is responsible for the distribution of its works”*.

252 AFCEN publishes “rules” for use in the French Nuclear Industry. It is made up of members from within the French nuclear industry and a small number of independents. The entire executive committee and editorial board is comprised of individuals from EDF

and AREVA. It is configured in a different manner than the ASME code development committees or the Eurocode committees. It does not have the participation of the French Nuclear Regulator ASN within its structure as a formal approver. AFCEN is not a standardisation body and its publications do not have standard status in the same way that ASME or the Eurocodes do. However, where possible they are based on existing standards, with the provision of additional information or clarifying options. Where no appropriate standard exists or modification thereto would be excessive, the AFCEN codes retain the status of stand-alone specifications. The Multinational Design Evaluation Programme (MDEP) considers AFCEN and ASME to be at the same level as a Standard Development Organisations (SDO) and is consulting AFCEN within the CORDEL Working Group for International Standardisation of Nuclear Reactor Design.

253 A review of the ETC-C AFCEN 2010 and the UK companion document (Ref. TSC55) has identified a number of areas where there are concerns. These are outlined below.

Part 0

The key findings are that the document contains:

- Lack of independent review of the code.
- There are a large number of references to French standards, with translations not provided.
- Loose references to “equivalent standards”.
- There are no references to national annexes to some standards such as EN13670.

Part 1 Design

The key findings are that the document contains:

- Errors in Formulas.
- Lack of Clarity/ ambiguity in text.
- Inconsistency with other sections of the code.
- Inconsistency with UK National annexe.
- Lack of guidance to designers on seismic design.
- Revisions of supporting documents unclear.
- Lack of guidance on choice of Eurocode value when no recommended value is available.
- Justification for revised liner stress limits is required.

Part 2 Construction

The review of Part 2 is still ongoing, as the mapping document for a number of sections has not yet been provided. However the interim key findings are that the document contains:

- Insufficient information to be the basis of a clear construction specification.
- Links to French ministerial standards are of no relevance to the UK.

- Lack of clarity of the intention to demonstrate the equivalence of French standards to other national standards .
- Lack of clarity over approval of modifications or adaptions to the specification.
- Lack of clarity over how demonstration of equivalence would be achieved.

Part 3 Leak and Resistance Testing

- The acceptance criteria require further clarification. This is currently linked to Assessment Finding **AF-UKEPR-CE-38**.

254 I have provided detailed comments to EDF and AREVA in Refs ND25 and ND26.

255 I have concluded that complementary clarification / justification are necessary to confirm the latest version of the ETC-C, ETC-C AFCEN 2010 along with the UK companion document as suitable for the design of the civil structures for a UK EPR. This is GDA Issue **GI-UKEPR-CE-02.**

256 There is a further finding that the licensee will need to re-visit all calculations undertaken to ETC-C Rev B which are to be used directly to support the design of the UK EPR to confirm that they are compliant against ETC-C AFCEN 2010 and the UK companion document once they have been agreed as suitable for use in the UK (Assessment Finding **AF-UKEPR-CE-07**).

4.3.3.6.3 ETC-C AFCEN 2010 UK Companion Document Assessment

257 The UK companion document (Rev A) is lacking in of information and requires complementary clarifications to confirm it as suitable for the design of the civil structures for a UK EPR. This is part of GDA Issue **GI-UKEPR-CE-02**.

4.3.3.7 Reliability of ETC-C

258 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. For the seismic case, the response was clear, as there is a clear progression of hazard with frequency, which can be defined, and the convolution of the hazard with the fragility is relatively simple to give a compound risk. However for the overpressure case, there are two complications. Firstly, there is not a continuous progression of increased pressure from fault scenarios with decreased frequency, rather there are a series of discrete fault scenarios which have attendant frequencies associated with them. Secondly, the claims made in the overall safety case assume a complete loss of containment as a mitigation for certain fault scenarios, typically those involving loss of the containment heat removal system. The convolution of the hazard and the fragility was therefore done in a simplified manner to predict the required reliability of the containment. The initial submission did not achieve a sufficient level of clarity on this, and we wrote to EDF and AREVA (Ref. ND21). This letter also discussed a number of concerns over the other two key submissions associated with RO-UKEPR-037, Refs EA76 and EA77.

259 Ref. EA77 discussed the reliability of the containment under seismic loads. The methodology used for this assessment is one developed by EPRI (Refs OD24 and

OD25). This methodology, whilst containing a number of simplifications has been accepted as providing an acceptable basis to evaluate the seismic reliability. These documents were developed partly under funding from the Nuclear Regulatory Commission (United States of America) US NRC and are used widely for this purpose.

260 The deterministic design for the containment contains a load case which encompasses the design basis earthquake and a simultaneous overpressure (SBLOCA + Earthquake, Load case 16, ETC-C AFCEN Table 1.3.3-2). The development of a surge line break LOCA from a (design basis) seismic event as an initiator is specifically designed out. This load case is therefore an onerous one. The corollary to this is that when considering seismic loading alone, there is considerable margin beyond the design basis event.

261 The development of the beyond design basis reliability against overpressure has a number of aspects:

- Overall methodology;
- Analysis;
- Choice of Critical Sections; and
- Development of fragility curves.

The initial review of Ref. EA76 was a complex process as it draws on a considerable number of supporting references. Ref. ND21 provided the formal feedback on the RO-UKEPR-037 submission. A series of technical meetings were held in late 2010 to clarify these points. The key areas of concern raised are detailed in the following paragraphs.

262 The response to RO-UKEPR-037 failed to address a number of the topic areas listed in the original request, including reliability of the engineer using the code, partial safety factors in ETC-C, construction quality, benchmarking and loading probability.

263 The use of the EPRI methodology (Refs OD24 and OD25) for the overpressure fragility has not been fully justified. The key concern is that the distribution of uncertainties for a seismic load is highly skewed by the hazard, which is not the case for the overpressure case. The uncertainties for the seismic case and the overpressure case originate from the loading side and the resistance side. For the seismic case, the uncertainties on the loading (i.e. hazard) dominate, and the EPRI methodologies used therefore tend not to focus unduly on other sources of uncertainty such as on the resistance side. For the overpressure case, the converse is true, where in the examples given, the loading values are a given, and the only uncertainty is the resistance. The simplifications in the EPRI approach therefore need to be justified when applied to the overpressure case.

264 It is unclear how EDF and AREVA have arrived at the various model and materials uncertainties used in their calculations. Some information is given in Ref. EA76 but not for the complete range of failure modes and locations. For example, a 2% coefficient of variation is given for some of the material properties, which is a very low value. It is stated in some of the documentation that typical figures for material variability have been taken from test data obtained during construction of one of the French PWRs. However, this approach needs to be viewed with caution as it may contain only some of the sources of variability. For example, if material (e.g. pre-stressing steel) has been obtained from only one steel supplier and from a limited number of production batches, then the variability may appear to be low. However, if different sources of supply are taken into account and differing approaches to quality control (as is possible within the various EU countries and outside) then the overall coefficient may be much greater than 2%. Further justification is required for the various sources of uncertainty that have been used in the calculations, and these need to be related to the quality control methods.

265 There is a lack of clarity over which design has been analysed for the overpressure case. In some areas it is described as Flamanville, in others Olkiluoto.

266 There are a number of concerns over the analysis undertaken for the beyond design basis studies including, the non linear material models, the treatment of boundaries and the apparent differences in geometry from the reference design.

267 Both the seismic and the over pressure studies ignored through shear, this may be reasonable for the seismic study but for the over pressure study it is likely that large shears will be developed at the base of the wall where it is restrained by the raft. In addition through shears around the access penetrations may be critical. It should also be noted that methods of predicting through shear capacity have high levels of variation and so the uncertainty for the through shear models will be large. This may mean they govern at high levels of confidence even if they do not govern for median capacities.

268 Revised responses to RO-UKEPR-037 have been received (Refs EA157 to EA165). These have not arrived in sufficient time to allow a meaningful assessment in the timeframe of GDA Step 4. Completion of this review is the subject of GDA Issue **GI-UKEPR-CE-05**.

4.3.3.8 Summary

269 The following key points have emerged from the assessment:

- ETC-C is restricted to application for the design and construction of EPR type nuclear power plant structures. It is not equivalent to a standard design code and would not be suitable to be used by a designer unfamiliar with such plant.
- A GDA Issue has been raised requiring further justification for the use of the ETC-C AFCEN 2010 version and the UK Companion Document for the design, construction and testing of the UK EPR civil works structures (GDA Issue **GI-UKEPR-CE-02**).
- A GDA Issue has been raised on the reliability of ETC-C AFCEN for design of the UK EPR (GDA Issue **GI-UKEPR-CE-05**).

270 The following assessment findings have been raised:

> ***AF-UKEPR-CE-06:*** *The licensee shall undertake any necessary fire tests on reinforced concrete walls using the actual materials to be used in the construction in accordance with the requirements of EN1992-1.2. This shall be undertaken ahead of the placement of Nuclear Island safety-related concrete. The higher strength concrete to which this test applies is only used in the inner containment. The test should be undertaken at this stage to ensure that the design meets its safety functional requirements before being fully implemented.*

> ***AF-UKEPR-CE-07:*** *The licensee shall re-visit all calculations undertaken to ETC-C Rev B which are to be used directly to support the design of the UK EPR to confirm that they are compliant against ETC-C AFCEN 2010 and the UK companion document. This shall be undertaken ahead of the placement of any structural concrete which the calculations relate to.*

4.3.4 Finite Element Code

4.3.4.1 Scope

271 As part of the design of civil structures there has been a considerable volume of analysis of the behaviour of structures under the postulated loading scenarios. The more complex analyses are focused around the seismic loading, aircraft impact, and pressure transient.

272 It is essential that there is a high level of confidence in the modelling used, the analyses performed and the outputs used. Those areas which were scrutinised in detail are:

- Appropriate use of analysis techniques.
- Validation and Verification of Codes.
- Application of Codes to structures.
    - Meshing.
    - Use of simplifications/ superelements.
    - Material models.
- Idealisation of Loadings.
- Validation of predicted responses.
- Output generation for use in design.

273 The key steps during the Step 3 and 4 assessment were:

- Identify structures/ loadcases where analysis techniques have been applied.
- Identify analysis methods and codes used.
- Review the above, and select subset for further examination.
- Identify key process documents and review against good practice, i.e. National Agency for Finite Element Methdods (NAFEMS).
- Perform selected deep slice review of modelling based on findings earlier to include:
    - Model idealisation.
    - Model testing.
    - Material modelling.
    - Loading idealisation.
    - Validation of response.
    - Output management.

4.3.4.2 Standards

274 The key SAPs which are applicable to this area are as follows.

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.		

Engineering principles: civil engineering: structural analysis and model testing	**Structural analysis and model testing**	**ECE.12**
Structural analysis or model testing should be carried out to support the design and should demonstrate that the structure can fulfil its safety functional requirements over the lifetime of the facility.		

Engineering principles: civil engineering: structural analysis and model testing	**Validation of methods**	**ECE.15**
Where analyses have been carried out on civil structures to derive static and dynamic structural loadings for the design, the methods used should be adequately validated.		

275 In addition, guidance provided by NAFEMS (Ref. OD7) is used for supplementary guidance.

276 One of the common issues raised with all software is that of Verification and Validation. Within the SAPs, there is much use of the terms Verification and Validation. They are defined as follows in the SAPs:

- Verification (in the context of computer codes) is the demonstration that the results calculations are the same as those intended by the authors of the code.
- Validation (in the context of computer codes) is the demonstration that the code and numerical model are appropriate for specific application intended.

277 EDF and AREVA however have a different approach to this subject which was clarified in Ref. EA4, and is repeated below for information. They introduce the concept of 'qualification'.

Verification and Validation

These two terms are considered together as EDF and AREVA consider "verification" to be an integral part of the "validation" process. Validation occurs before the receipt of the application by the EDF user organisation. The objective of this process is defined by EDF as: *"to review the design of the application, using a process by which, through testing, the satisfactory design of the application and its fulfillment of its design specifications can be confirmed."*

'Verification' is a process undertaken by the software developer by which the software undergoes a series of checks and tests to ensure that it performs the functions for which it was designed, i.e. that it conforms to the design specification. The developer is expected to provide a Software Quality Plan describing the verification process and may be asked to provide a summary report of the verification process.

'Validation' is the process by which the capability of the application to perform its intended purpose is confirmed. This process is also undertaken by the software developer, but is likely to involve the receiving organization within EDF. The developer runs tests defined in the project specification for the application. The results of these tests are included in a final validation report (this may also be known as a 'factory receipt').

Qualification

'Qualification' of a scientific or technical application is defined by the French standard AFNOR Z61-102 as: *"a procedure by which a competent authority verifies that, after the validation phase, the application satisfies the specification which it was designed to fulfill, in accordance with the application's Quality Plan."*

The initial part of the qualification phase is the receipt and installation of the application, during which the compatibility with IT systems is checked, and the suitability of the User Manual and all other supporting documentation is confirmed.

The main part of the qualification phase is the demonstration that the application is suitable for use in the domain for which it is intended to be used. This domain, known as the 'domain of qualification' is defined precisely in the qualification report: It includes parameters such as: the technical environment and physical domain in which the code is to be used, the limits of application, the calculation methods to be used, etc. Various methods can be used to qualify an application:

- Additional tests (beyond those undertaken during validation) to confirm that the application produces satisfactory results (for example when used in a specific methodology, or when used in conjunction with other qualified software applications).
- comparison with results of laboratory or in situ testing.
- comparison with results from previously qualified applications.
- use of expert opinion, or experience of users of the application (in the same domain of qualification).
- certification by a suitable and reputable organization.

Once an application has been qualified for use in a specified domain, it can be used to undertake studies and calculations in that domain without further justification. However, if the application is to be used in a domain other than that for which it is qualified, it must be re-qualified for use in the new domain, provided that the effort necessary for re-qualification is not disproportionate to the importance of the study in question.

278 It is seen that the two approaches should achieve the same ends, however there is a need to be careful when interpreting EDF and AREVA documents that terminological issues do not cloud the judgement over the adequacy of the processes applied to the analysis codes used.

4.3.4.3 Overview of Codes Used

279 A wide range of Finite Element Codes and related software has been used in the UK EPR design. Figure 3 provides an overview of the most important codes used. The sections below summarise the types of code used and their use in the design. During the assessment process, there has been a continual discovery of further codes, code checkers, translators and manipulators. Table 1 provides an overview of the codes examined in GDA and their key function in the design.

280 ***Code ASTER*** (www.code-aster.org)

Code_ASTER is a software package for finite element analysis and numeric simulation in structural mechanics originally developed as an in-house application by EDF. It was released as free software under the terms of the GNU General Public License, in October 2001. It is implemented through over 1,500,000 lines of source code, most of it in Fortran. The software is provided with about 2,000 validation and verification examples.

The documentation is extensive with more than 14,000 pages of user's manuals, theory manuals and qualification examples, the vast majority of which is in French. It is used for the finite element modelling of the main Nuclear Island structures and can be linked to the code MISS 3D which provides specific soil-structure interaction capability.

281 ***Code ProMISS3D***

ProMISS3D is a calculation code developed by the Central School of Paris (École Centrale de Paris), and is a development of MISS3D. It is based on the Boundary Element Method (BEM). It is a modular dynamic 'soil-structure-fluid' interaction program and can be used with shapes of all kinds, heterogeneous soils, and multiple deep or surface foundations. The MISS modules handle two-dimensional or three-dimensional analysis, including for works that are infinite in one direction

282 ***Code EUROPLEXUS*** (europlexus.jrc.ec.europa.eu/)

EUROPLEXUS is general Finite Element software for the non-linear dynamic analysis of Fluid-Structure systems subjected to fast transient dynamic loading such as:

- Explosions in enclosures.
- Study of shocks and impacts of projectiles on structures.
- Analysis of pipelines in transient mode.
- Safety evaluations of complex Fluid-Structure systems under accidental situations.

EUROPLEXUS is jointly developed by the French Atomic Energy Commission (CEA), the Joint Research Centre (JRC) of the European Community, EDF, ONERA (Office Nationale de Recherche en Aérospatiale) and SAMTECH.

283 ***Code COBEF*** (www.coyne-et-bellier.fr/en/dun/dsc/logiciels.html)

COBEF is a finite-element analysis program for static or dynamic (spectral, temporal) calculation of linear or non-linear elasticity problems. It is a code which has been developed over many years by Coyne et Bellier for their own internal use. The numerical analysis can be two-dimensional with plane stress, plane strain, or axisymmetrical, or can be three-dimensional. COBEF uses a wide variety of finite elements (springs, bars, beams, surface and volume elements, membranes, shells, and 1D or 2D joints). These elements can be linear, quadratic, or incomplete, and are all mutually compatible. The program can handle distributed and thermal loads, nodal forces, and hydrostatic or other pressures.

284 ***Code PRECONT***

PRECONT is a Coyne et Bellier Code which calculates the distribution of loads along defined tendon geometries and then discretizes the forces such that they can be applied to the ANSYS model of the containment. It models the effects of friction, creep, shrinkage, wobble and draw-in.

285 ***Code ASTHER/HERAST***

ASTHER/HERAST is an interface code which translates the output from ASTER into a form which code HERCULE can read. It has been developed in-house by IOSIS.

286 ***Code ANSYS*** (www.ansys.com)

ANSYS Mechanical and ANSYS Multiphysics software are non exportable analysis tools incorporating pre-processing (geometry creation, meshing), solver and post-processing modules in a graphical user interface. These are general-purpose finite element modeling packages for numerically solving mechanical problems, including static/dynamic structural analysis (both linear and non-linear), heat transfer and fluid problems, as well as acoustic and electro-magnetic problems. ANSYS is developed by ANSYS Inc, and has been available for many years as commercial software. ANSYS is used to model the containment structure under all internal and external load cases, including accident transients.

287 ***Code SYSTUS*** (www.esi-group.com/products/multiphysics/systus)

SYSTUS is a multiphysics simulation software. SYSTUS is an implicit code which covers fields as diverse as civil and mechanical engineering, energy and transportation. Systus is used to aid the design of the steel containment liner. The models of the liner are subjected to displacements calculated from the ANSYS models.

288 ***Code FERRAIL***

FERRAIL is a Coyne et Bellier in-house code which is used to calculate areas of reinforcing steel required according to rules in the ETC-C code, taking loads from either ANSYS or COBEF as primary input.

289 ***Code HERCULE***

Code HERCULE is a general purpose finite element code and code checker developed by SOCOTEC in France, and used by IOSIS on some of the Nuclear Island structures.

290 ***Code NASTRAN (***www.mscsoftware.com)

NASTRAN is a long established finite element code originally developed by NASA and now available through msc software (www.mscsoftware.com). It is capable of the analysis of linear, non linear, static and dynamic problems.

291 ***Code SOFISTIK*** (www.sofstik.com)

SOFISTIK is a general purpose finite element code available via SOFISTIK AG. It is capable of the analysis of linear, non linear, static and dynamic problems, and is commonly used in the design and analysis of large building and bridge structures.

292 ***Code HFERCOQ***

HFERCOQ is an in-house code checker developed by IOSIS for use with the finite element code HERCULE

293 ***Code CASTEM*** **(GIBI)**

CASTEM is a general purpose finite element code, and GIBI is its mesh generator. GIBI has been used to generate the mesh for the whole nuclear island model. CASTEM however has not been used for any finite element calculations.

294 ***Code HER 2 COB***

HER 2 COB is a translation software which allows for the conversion of HERCULE input files into a format which COBEF can utilise.

4.3.4.4 Overall Strategy

295 The codes listed above fall into two broad categories. Firstly, there are those which we have regulatory experience of and have examined on a number of occasions previously. The second type are those which we have no previous knowledge of. The approach adopted has been quite different for the two types.

296 Those codes which we have detailed prior knowledge of are:

- ANSYS
- NASTRAN

These are well established codes and are available with extensive suites of validation and verification documentation. In addition, they have been subjected to extensive regulatory scrutiny in the UK and elsewhere for many years. ANSYS design analysis software is created within a quality system with ISO 9001 certification, the internationally accepted quality standard. Product development, testing, maintenance and support processes also meet the United States Nuclear Regulatory Commission's quality requirements. NASTRAN is similarly well supported by quality systems and certification. Both these codes are considered adequate for the purposes they have been used in GDA.

Code SYSTUS has been used previously in the UK for the Sizewell B project, and we have some limited knowledge of it. The assessment has been therefore less in-depth than for those codes which we know little.

297 Those codes of which we have little if any previous knowledge are listed below:

- Code ASTER
- Code ASTHER/HERAST
- Code ProMISS3D
- Code Europlexus
- Code SOFISTIK
- Code Precont
- Code Ferrail
- Code HERCULE (inc HFERCOQ)
- Code CASTEM (GIBI)

For these codes a detailed review of their development, verification and validation arrangements has been made. The levels and depth of the review is commensurate with the relative importance of their use and the safety claims on the items analysed using the software.

4.3.4.5 Assessment

298 A meeting was held with EDF and AREVA on 25th April 2009 to discuss the overall use of finite element codes in the design of civil engineering structures. This was useful in confirming the codes used and the scope of their application. In addition, it gave an initial impression of the nature of the support arrangements for each code in terms of verification and validation. An overview of the capabilities and pedigree of the software was also provided. During the remainder of the more detailed assessment of the design of the safety-related nuclear structures it became clear that the list of software used was somewhat larger than initially anticipated. As the assessment has progressed through Step 3 and 4, the number of software packages identified has increased steadily. The following sections discuss the individual codes and their assessments.

4.3.4.5.1 Code ASTER

299 The scope of the evaluation of Code ASTER included the following:

- Establish the full extent of Code ASTER used in the analysis of civil structures of the UK EPR design.
- Understand the development history and the development process of Code ASTER.
- Identify, review and assess the Quality Assurance process documents and compare with current best international practice such as those recommended by NAFEMS and ASME.
- Examine verification and validation of Code ASTER and compare with current best international practice such as those recommended by NAFEMS and ASME.
- Assess a reasonable number of samples of Code ASTER theory, benchmarking and validation documents.

300 In addition, a series of face to face meetings were held, including the development team for Code ASTER.

301 The full extent of Code ASTER used and its application in the civil and structural works of the EPR is as follows:

- Linear elastic analysis of all the buildings on the common raft foundation.
- Seismic analysis of the primary structures is carried out using linear elastic response spectrum analysis method. Floor response spectra were obtained by the linear response history analysis method in the time domain or the linear transfer function method in the frequency domain.

302 Ref. TSC9 contains a thorough review of CODE ASTER, the key points of which are repeated below.

303 Code ASTER has had a 20 year history of development by a dedicated team of 15 to 20 computational mechanics specialist research engineers having PhD degrees at the R & D division of EDF. A 20 year development history is considered appropriate for a code of this nature to grow its capabilities, for bug-fixing and for the required verification and validation work to be carried out.

304 Code ASTER has been used for analysis of the civil works of the Flamanville 3 in France and Olkiluoto 3 in Finland, both currently being constructed, in addition to several nuclear plants in France since 1996. It does have a history of application in computational mechanics analysis of civil structures in nuclear safety-related plants.

305 Code ASTER has been reviewed by two independent third parties. In addition, Code ASTER has had some degree of regulatory oversight by the French Nuclear Safety Authority (ASN) on its Quality Assurance process and on the analysis results obtained using Code ASTER on nuclear plant structures.

306 The development process and the development activities of Code ASTER are well organised and well managed. The verification, validation and qualification processes of Code ASTER are broadly consistent with current best international practice as recommended by NAFEMS and ASME. The EDF definition of verification is equivalent to “code verification” as defined by NAFEMS and ASME – solving the equations right. The validation process and part of the qualification process of Code ASTER are equivalent to the validation process as recommended by NAFEMS and ASME – solving the right equations.

307 The Quality Assurance process documents of Code ASTER sampled are good quality documents. The various code development, maintenance, verification and validation, and documentation processes are well organised and well managed with clearly defined activity process flow charts and quality assurance responsibilities for various activity participants.

308 Extracts of selected Code ASTER theory manuals (reference documents) show that the finite element formulations in Code ASTER are based on rigorous mechanics principles and matrix mathematics operations.

309 Assessment of selected validation documents of the selected functions of Code ASTER for detailed examination suggests that Code ASTER has gone through a rigorous and comprehensive validation process. For all validation problems, the solutions obtained from Code ASTER closely match the benchmarking (reference) solutions obtained analytically or by another finite element code.

310 In summary, Code ASTER is seen as suitable for the analysis of the EPR civil structures and is compliant with the appropriate SAPs.

4.3.4.5.2 Code ProMISS3D

311 The scope of the evaluation of Code MISS 3D included the following:

- Establish full extent of the codes used and their applications into the design.
- Review the development of the codes focussing on the following aspects:
    - Development history.
    - Development process.
    - Third party reviews.
    - Application history.
    - Regulatory oversight/ reviews.
- Review any benchmarking claims.
- Review extent of validation and verification undertaken.
- Establish limitations on usage in the context of the applications in question:
    - Structure type.
    - Model size.
    - Loading Function.
    - Non linear capability.
    - Physical behaviour.
    - Material models.

312 Ref. TSC8 contains a thorough review of the code, the key points of which are repeated below

313 Developed and maintained by the Ecole Centrale Paris (Aubry and Clouteau, 1992), the code was designed originally to perform 3D soil-structure interaction analyses using the sub-structure approach in the linear elastic or viscoelastic domain. It also incorporates the boundary element method (BEM). An interface exists between MISS3D and

Code_ASTER for soil-structure interaction analyses. Several conference and journal papers have been identified (e.g. Clouteau et al, 1998; Pitilakis et al, 2006) which describe the various aspects of the code and present comparative studies with other codes, for example FLUSH (Lysmer, 1975). A general overview of the program can be located at http://www.mssmat.ecp.fr/structures/missuk.html#DD.

314 MISS3D uses a boundary element method approach. The BEM is a numerical computational method for analysing dynamic soil-structure interaction. It is different from finite element analysis because only the surface behaviour of the model is computed, instead of computing the behaviour of the whole of the interior of the model. Firstly, the area being modelled is divided into sub-domains, for example: the structure and the ground. Rather than solving these partial differential equations throughout the sub-domain a solution is obtained to the integral equation defined on the boundaries of the sub-domain. The boundary element method produces an exact solution on the boundary for linear elastic materials. It is also possible to use the solution to the integral equation to extract the solutions at any desired point within the subdomain.

315 Green's functions are typically used to solve the integral equation subject to specific boundary conditions. In particular, they can be used to give exact solutions to unbounded or semi-infinite regions, for homogeneous or stratified soil profiles. The boundary element method can be used in conjunction with other methods, in particular with the finite element method. Boundary behaviour output from a finite element analysis can be input as the boundary conditions of a sub-domain in the boundary element analysis. The results from the boundary element analysis can then be incorporated into the finite element analysis. Code_ASTER and ProMISS3D work together in this way.

316 Solids may be modelled as homogenous or stratified, the latter being accommodated using the principle of superposition. If modelled as stratified, the data required is number of layers, height, material types of different layers and the material properties of the underlying half-space (bedrock). Both infinite homogeneous and stratified elastic sub-domains are modelled using the boundary element method and Green's functions to calculate an exact solution. Although ProMISS3D has a capability for modelling horizontal stratification (each layer having uniform values), sloping ground surfaces or sloping subsurface layer interfaces are not allowed. In the case of embedded foundations, the soil must be layered such that the base of the foundation coincides with the base of the top layer of soil.

317 The analysis is run in three stages. Preliminary analysis in Code_ASTER finds the dynamic modes of the structure and the static modes of the interface. This data is transferred, along with the foundation surface element mesh, to ProMISS3D via the programme "gtaster" for analysis using the Boundary Element Method (BEM). ProMISS3D then solves the integral equation over the interfaces of the sub-domains modelled (soil and structure) and returns the soil impedances via the programme "ptaster" to Code_ASTER. Post-processing in Code_ASTER is then carried out. The distribution of the global impedances for the foundation into discrete springs and dashpots beneath the base of the structure is done via a sub-routine RIGI_PARASOL.

318 RIGI_PARASOL works as follows: each of the 6 values of global rigidity of the soil, corresponding to the degrees of freedom of the rigid body, are distributed to each node on the underside of the foundation using a distribution function. The distribution function takes into account the surface area related to each surface node and the distance of each surface node from the centre of gravity of the structure. The practical limitations and application of this approach is discussed in Section 4.2.5.3.3.2.

319 The majority of the documentation provided on the internal verification of ProMISS3D and its use within Code_ASTER are either limited extracts of studies or non-regression tests. From these it has been difficult fully to assess the verification and benchmarking work undertaken and establish a clear position on the appropriateness of Code_ASTER – ProMISS3D for the analysis of seismic SSI for nuclear new build facilities. The non-regression tests provide some confidence that changes to the code are not inducing inadvertent changes in the operation of ProMISS3D. However, even with these the nature of the documents supplied has made it difficult to assess the true nature and purpose of the tests. A good example of this is the non-regression tests which refer to CLASSI – POUX in addition to Code_ASTER – ProMISS3D. Without undertaking a more complete review of these codes as well, it is difficult to be absolute over the merit in the comparison.

320 However, the external verification and benchmarking tests are of more use in assessing the technical performance of the code. The external verification tests compare ProMISS3D results with reference impedances from simple closed-form solutions by Seiffert and Cevaer. The errors quoted between ProMISS3D and the reference impedances are appreciable and regularly exceed 5% by some margin. I consider a target error value of 5% would represent a reasonable upper bound for the simple examples considered in these verifications. The benchmarking of Code_ASTER – ProMISS3D and Seismic Analysis of Soil Structure Ineteraction (SASSI) indicates reasonable agreement between response spectra. However, significant errors were apparent between the impedances provided by the packages at relatively low frequencies. This is of concern as the fundamental frequency of the soil columns at UK sites is also likely to be relatively low and the impedances are the key results which are taken forward into the structural FE modelling. A further concern is that SASSI also uses boundary element techniques to represent the soil. A more robust benchmarking would be provided by comparing Code_ASTER – ProMISS3D against a time domain finite element package. SASSI is a well know code in this particular area and has been benchmarked many times against standard solutions.

321 The use of the Boundary Element Method in principle for hard sites is accepted; however its use for softer sites, typical of those in the UK as a primary analysis tool is questionable.

322 The BEM has the following limitations:

- BEM application is restricted to linear materials. The vast majority of BEM formulations are based on fundamental elastodynamic solutions of wave propagation problems. Therefore the materials involved are assumed to behave in an elastic or visco-elastic manner.
- BEM application is restricted to simple geometries and ground conditions. The fundamental solutions employed by the BEM involve very simple geometries and typically homogeneous ground conditions.
- BEM requires complex mathematical formulations and it does not provide a physical insight of the SSI problem. The BEM formulation requires very complex mathematical operations which are difficult to be understood by “non BEM experts”. Inevitably in engineering practice the method is often used as “black box” without appreciating the underlying assumptions.

323 ProMISS 3D is considered suitable for use on hard sites such as Flamanville where the soil is high strength rock which will suffer little or no degradation under the design earthquake.

324 The use of ProMISS3D in the UK for sites other than those considered hard would require considerably more justification, and the consideration of the non linear behaviour of soils in a more detailed fashion. This is Assessment Finding **AF-UKEPR-CE-08**.

4.3.4.5.2.1 Code EUROPLEXUS

325 EUROPLEXUS has been examined in a large degree of detail. The verification and validation approaches for the code have been discussed extensively, and a large number of documents sampled. Ref. TSC8 contains a thorough review of the code, the key points of which are repeated below.

326 The following paragraphs summarise the findings.

327 EUROPLEXUS is an established, commercially available, structural analysis package that is used by some large companies and research institutions around Europe in the power generation and aerospace industries. EUROPLEXUS has been used for a variety of types of analysis in the power generation and aerospace industries.

328 The development process and the development activities of EUROPLEXUS are organised and managed by the code developers CEA, JRC and SAMTECH. Although there is only limited information concerning the development process and activities, what is known appears to show that it is well managed. CEA and JRC are large and reputable research institutions. SAMTECH appears to be a professional and competent software company.

329 Although there has not been any direct oversight or review of EUROPLEXUS by a regulatory authority, the French regulatory body (ASN) has accepted designs where EUROPLEXUS has been used. The software has been shown to satisfy the requirements of ASN in terms of its qualification.

330 The activities undertaken using EUROPLEXUS in the design of the UK EPR uses many generic solution and calculation methods within EUROPLEXUS that can be found in many other finite element codes. Most of the functions used in the EUROPLEXUS analyses are reasonably standard features of most finite element codes. The solution method used in EUROPLEXUS (explicit time integration) is a suitable method for the analyses undertaken.

331 The GLRC DAMA material model developed by EDF R&D is considered a suitable method for representing the overall bending, membrane and shear behaviour of the structures examined using EUROPLEXUS. In addition, the use of the sub-domains methodology is considered acceptable.

332 The verification and validation processes have been examined in detail and the following key points found.

- The EDF definition of “verification” is consistent and equivalent to “code verification” as defined by ASCE/NAFEMS
- EDF considers that the “verification” process is an integral part of the “validation” process.
- The remaining part of the “validation” process and part of the “qualification” process appear to be consistent and equivalent to the validation process as defined by HSE/NII and ASME/NAFEMS.

- The EDF “qualification” process includes an additional element of certification by “a competent authority” and originates from a French National Standard AFNOR Z61-102.
- Available evidence suggests that EDF have adopted a self “qualification” or self certification approach for qualification of EUROPLEXUS.
- The EDF definition of “verification”, “validation” and “qualification” processes broadly meets the SAPs standard ECE.15, sections 294 and 295 and are broadly consistent with current best international practice as recommended by NAFEMS and ASME.

333 In summary, Europlexus is considered to be a suitable code for the purposes it has been used for in the EPR design.

4.3.4.5.3 Code Systus

334 Code SYSTUS is marketed by ESI, which actively markets a series of software applications for use in a range of industries. Systus itself has been available for over 20 years and has a large industrial base of users. Comprehensive user and theory manuals are available and have been provided.

335 It should be noted that this code was used as part of the Sizewell B project, for analysis of the RPV, and therefore has some previous pedigree and acceptance within the UK regulatory regime.

336 The SYSTUS User Manual submitted in partial response to TQ-EPR-1025 has been reviewed. The User Manual is thorough, and indicates that the software is sophisticated and well-designed. Equally, the verification and validation examples provide a high degree of confidence in the capability of the software.

337 In summary, SYSTUS is considered to be a suitable code for the purposes it has been used for in the EPR design.

4.3.4.5.4 Code Precont and Ferrail

338 Initial discussions were held with Coyne et Bellier in Step 3 to gauge the extent and nature of the code and to judge the level of any future regulatory review. A subsequent inspection of the codes found that a clear verification and validation path existed, and that the software was included in the overall quality system operated by Coyne et Bellier which is ISO9000 certified. A review of the manipulation of the output from finite element analysis suggests that an appropriate approach is adopted in the code.

339 During Step 4, a more detailed review of the outputs from FERRAIL have shown that there are no concerns over its ability to calculate reinforcement areas (Ref. TSC2).

4.3.4.5.5 Code SOFISTIK

340 SOFiSTiK is a general finite element code for the analysis of the structural behaviour. It is developed and maintained by SOFiSTiK AG in Oberschleißheim, Germany (near Munich).

341 The SOFiSTiK finite element code is widely used for civil and structural applications and has a large user base. Starting in 1973, the software package was developed by several German civil engineers to support the design process in civil and structural engineering. The software has been continuously developed since 1981. In 1999, the merger of

individual engineering offices and a marketing company formed the SOFiSTiK AG Corporation.

342 SOFiSTiK can be used for a number of different linear, non-linear and static and dynamic civil and structural calculation problems. Currently, the analysis and modelling capabilities of SOFiSTiK include:

- Static and dynamic and linear and non-linear analysis for buildings and bridges.
- Reinforced concrete and steel structural analysis.
- Construction stage analysis in bridge and geotechnical engineering.
- Wind and seismic analysis.
- Geomechanics and tunnel design.
- Retaining wall design.
- Seepage, hydration and fire design.
- 3D tendon geometry.

343 SOFiSTiK AG also claims to be an active member of national and international associations and societies of industry and profession, including:

- National Agency for Finite Element Methods and Standards (NAFEMS).
- International Association for Bridge and Structural Engineering (IABSE).
- International Association for Computational Mechanics (IACM).
- German Association for Computational Mechanics (GACM).
- Verband Beratender Ingenieure (VBI) – the German Association of Consulting Engineers.

344 TQ-EPR-632 was raised to obtain information on any oversight/reviews by a national regulatory agency. In the full response to this TQ, EDF and AREVA confirmed that:

- It is not common practice in Germany for regulatory bodies to carry out oversights or reviews on software used for nuclear plant analyses.
- There has been no regulatory review of SOFiSTiK by the French Nuclear Safety Authority (ASN).

345 However, EDF and AREVA also stated that:

- The German regulatory bodies have always accepted SOFiSTiK.
- The Finnish regulatory body STUK has accepted SOFiSTiK analyses for OL-3 in Finland.
- ASN has accepted SOFiSTiK analyses for Flamanville 3 in France.
- EDF and AREVA have referred to the list of nuclear power plants analysed using SOFiSTiK in their response to TQ-EPR-633 and the fact that other German companies have used SOFiSTiK in the design of nuclear power plants.

346 The SOFiSTiK Verification Manual, Version 12.2, 2009 was supplied by EDF and AREVA and is also available on the main SOFiSTiK website. This Verification Manual notes that the different tasks covered by SOFiSTiK are so large that it is not possible to validate all

specific features with known reference solutions. However, it mentions that there are a few sources of verification and validation:

- Internal verification examples maintained by the code programmers, but these are not publicly available.
- Examples given in the SOFiSTiK manual, which show the general behaviour of program and provide expected or approved results for comparison.
- Presentations given at the annual SOFiSTiK user meetings (since 1988) showing the practical usage of the software within a wide range of applications and the scientific background.
- Externally established examples such as NAFEMS benchmark tests, a number of which are included in the Verification Manual.

347 The externally established examples given in the Verification Manual consist of 35 simple verification tests, all of which are standard NAFEMS benchmark examples. The modelling and analysis aspects that these 35 tests cover include:

- Membrane, plate, shell and beam elements.
- Point loads, shear loads, distributed loads, gravity loading.
- Temperature loading.
- Axisymmetric modelling.
- Linear static solutions.
- Non-linear material, quasi-static solutions with perfectly plastic and isotropic hardening behaviour.

348 Each test involves only a small number of elements with relatively simple loadings. Therefore, NAFEMS provides closed form solutions to these tests. For each test, the documentation of the test includes a description of the test model, the loading, the modelling aspect being tested and comparisons of the SOFiSTiK results with the closed form NAFEMS solution. In all cases, the SOFiSTiK results are almost identical to the NAFEMS benchmark solution.

349 These tests verify the shell element behaviour, distributed loading and general non-linear material behaviour. While these NAFEMS benchmark tests are not comprehensive (i.e. they do not test every aspect of SOFiSTiK), they demonstrate a reasonable level of rigorous Code Verification testing.

350 Based on the documents reviewed, the following conclusions on SOFiSTiK have been drawn:

- SOFiSTiK is a well-established, commercially available, structural analysis package that is used by a large number of users and companies around the world.
- The development process and the development activities of SOFiSTiK are well organised and managed by SOFiSTiK AG.
- Overall, SOFiSTiK AG appears to be a professional and competent software company.
- SOFiSTiK has been used extensively for a variety of types of analysis in the Nuclear Industry over a long period of time.

- Although there has not been any direct oversight or review of SOFiSTiK by a regulatory authority, the German, Finnish and French regulatory bodies have accepted designs where SOFiSTiK has been used.
- The available NAFEMS benchmark tests, while not comprehensive, demonstrate a reasonable level of rigorous Code Verification testing.
- Published papers demonstrate the validation of SOFiSTiK analyses against experimental test results.
- Published papers demonstrate the benchmarking of SOFiSTiK analyses against other non-linear analysis methods.
- Sufficient verification and validation examples have been reviewed to establish a high level of confidence that SOFiSTiK is a suitable code for the analysis of civil structures of nuclear safety-related plants.

4.3.4.5.5.1 Code HFERCOQ

351 The scope of the evaluation of Code HFERCOQ is as follows:

- establish the full extent of the use of HFERCOQ in the post-processing of finite element analyses of civil structures of the UK EPR design and identify those functions which will be examined in more detail.
- understand and review the development history, development process, third party reviews, application history and any regulatory oversight/review of HFERCOQ.
- review and assessment of benchmarking.
- review and assessment of verification, validation, qualification and Quality Assurance processes.
- establish any limitations of HFERCOQ in the applications in question.

352 Ref. TSC24 contains a thorough review of the code, the key points of which are repeated below.

353 HFERCOQ was used for the structural design to ETC-C Revision B of the following structures for the reference design.

- Fuel Building (HK).
- Diesel Building (HD).
- Safeguard Auxiliary Building divisions 2 and 3 (HL 2 and 3).
- APC shell.
- Common Raft.

354 HFERCOQ was developed in 2005/6 by Sechaud & Metz, and in recent years by IOSIS, as a post-processor to the HERCULE FE analysis program, to calculate the shear reinforcement required in reinforced concrete plate or shell elements. HFERCOQ undertakes calculations in accordance with the BAEL or ETC-C (2006) depending on the option chosen. There were some errors in the English documentation, but these were corrected in the response to TQ-EPR-1390.

355 As it is proposed to design EPRs to ETC-C, only this code option within HFERCOQ has been considered. It should be noted, however, that HFERCOQ checks are currently to ETC-C (2006), and not to ETC-C (2010).

356 HFERCOQ has been tested and undergone quality control procedures within the company, as is necessary and essential for its use on IOSIS studies. IOSIS has quality control procedures to manage the development process, which include documentation, system of traceability, validation and qualification tests.

357 A detailed review of that the procedures described for the design to the ETC-C (2006) are in accordance with that document has been undertaken. It was found that some expressions were incorrect in the English versions of the overview document provided by EDF and AREVA but correct in the French version. EDF and AREVA have confirmed this in responses to TQ-EPR-1390.

358 In summary, it is considered that HFERCOQ is suitable for the design of reinforcement against the requirements of the ETC-C, although some revalidation will be needed to confirm compatibility with the ETC-C AFCEN version. This part of a more general assessment finding, Assessment Finding **AF-UKEPR-CE-07**.

4.3.4.5.6 Code HERCULE

359 The scope of the evaluation of Code HERCULE is as follows:

- Establish the full extent of the use of HERCULE in the analysis of civil structures of the UK EPR design and identify those functions which will be examined in more detail.
- Understand and review the development history, development process, third party reviews, application history and any regulatory oversight/review of HERCULE.
- Review and assessment of benchmarking.
- Review and assessment of verification, validation, qualification and Quality Assurance processes.
- Establish any limitations of HERCULE in the applications in question (e.g. structure type, model size, loading function, nonlinear capability, and material models).

360 Ref. TSC21 contains a thorough review of the code, the key points of which are repeated below.

361 HERCULE is used for finite element structural analysis of the following buildings of the EPR:

- The Common Raft.
- The Fuel Building (HK).
- Safeguard Auxiliary Building divisions 2 and 3 (HL 2 and 3).
- Diesel Building (HD).
- The APC Shell (non aircraft impact loading only).

362 Only the linear elastic analysis capability of HERCULE has been used. There is no documented evidence on the use of any nonlinear analysis capabilities of HERCULE being used in the EPR design.

363 HERCULE has been developed and maintained by SOCOTEC, a leading French company in construction inspection. However, SOCOTEC does not actively market HERCULE to promote its use in the civil and structural engineering profession. As a result, there is no information about HERCULE available in public domain sources. Assessment carried out in has been based on the very limited information and documentation provided by IOSIS (via EDF and AREVA). Being a user of HERCULE rather than its developer, IOSIS understandably is not in a position to provide comprehensive documentation and information similar to those EDF and AREVA have provided on Code ASTER. Nevertheless, on two critical aspects of this assessment – on the application history of HERCULE to civil and structural analysis of French and international nuclear power projects and on verification, validation and benchmarking, the documents provided are comprehensive and adequately detailed and are of an acceptable standard.

364 Available evidence suggests that SOCOTEC started development of HERCULE in the 1970s and has been continuously developing and maintaining it, although its development process is not clear due to the lack of information. Therefore, it has had a development and maintenance history sufficiently long for growing its capabilities and for bug fixes.

365 SECHAUD ET METZ (IOSIS since 2007) first used HERCULE on the Koeberg Nuclear Power Plant in South Africa in 1978. Since 1992, SECHAUD ET METZ (IOSIS since 2007) has been using HERCULE continuously on numerous French and International nuclear power projects. Notably in recent years, IOSIS used HERCULE on the Olkiluoto EPR in Finland, Flamanville 3 EPR in France and the Taishan EPR in China. Therefore, HERCULE does have an extended history of application in French and international nuclear power plants.

366 The French nuclear regulator ASN and its technical support institution IRSN have reviewed and approved studies performed using HERCULE for the French Flamanville 3 EPR project.

367 Verification, validation and benchmarking of HERCULE have been carried out and documented to acceptable standards. The scope of the verification, validation and benchmarking tests are comprehensive, covering a wide spectrum of solution procedures and its finite element library. On most test problems, several finite element models with different mesh densities have been created for the purpose of calculation verification. On each one of these test problems, a clear path of convergence to the reference solution has been demonstrated. Several benchmarking test problems against NASTRAN indicate that the performance of HERCULE thin shell elements is similar to that of NASTRAN.

4.3.4.5.7 Code COBEF

368 The scope of work of the evaluation of COBEF is as follows:

- Establish the full extent of the use of COBEF in the analysis of civil structures of the UK EPR design and identify those functions which will be examined in more detail.
- Understand and review the development history, development process, third party reviews, application history and any regulatory oversight/review of COBEF.
- Review and assessment of benchmarking.
- Review and assessment of verification, validation, qualification and Quality Assurance processes.

- Establish any limitations of COBEF in the applications in question (e.g. structure type, model size, loading function, nonlinear capability, and material models).

369 Ref. TSC22 contains a thorough review of the code, the key points of which are repeated below.

370 COBEF is a proprietary code owned and developed by COB. They were unwilling to allow removal of key documents from their premises, hence the review has been limited to time spent at their offices reviewing documentation and the responses to some TQs.

371 COBEF has been used for a variety of nuclear and for non-nuclear applications such as dams and general civil structures. Previous applications in nuclear power plants include Koeberg Nuclear Power Plant in South Africa, NPP series in France (Paluel 1 to 4), and Karun Nuclear Power Plant. For these projects, COBEF was used to model the Inner Containment. In addition, specific studies using COBEF were performed for various projects related to French PW 900 MW series (Chinon B1/B2, Dampierre, etc). For the design of the EPR, COBEF was first used in the development of the conceptual and the basic design studies of the EPR Inner Containment and Containment Internal Structures in the early phases of these studies since 1990.

372 COB have developed COBEF since the beginning of the 1970s for the purpose of supporting their design work on nuclear and non-nuclear civil engineering structures. The decision to develop their own FE code rather than purchase a commercial code was based on the need to adapt the capabilities to suit their requirements, and to have access to source code so that limitations could be understood and bugs could be fixed.

373 The User's Manual and all QA documents of COBEF are available for consultation in COB's office only. All are in French, although the titles of the QA documents and the titles of the chapters of the QA documents have been translated into English and listed in the COB design report.

374 Nevertheless, available evidence provide by COB suggests that the QA procedure implemented by COB on the COBEF code development, documentation, bug reporting and fixing, version management, and external QA audits is acceptable, although not as comprehensive and not up to the same standard as that of Code ASTER. The lack of a documented development process is evident.

375 A sample of the validation and verification test descriptions was inspected. The purpose was to check whether the types of elements and solution methods relevant to analysis of the EPR FA3 NAB have been validated appropriately.

376 The majority of the verification and validation examples chosen and their reference solutions are taken from the AFNOR guide (Ref. OD31). This publication gives test problems on beam elements, shell elements, spring elements and single degree-of-freedom oscillators.

377 The testing procedure is not written down explicitly. For each version of COBEF, COB use an automated process to run the suite of tests and produce a document listing the results. COB staff compare these results against results from previous versions of COBEF. If the results are deemed insignificantly different from those of the previous version of COBEF, the new version is accepted and signed off by the COBEF development manager as being suitable for use.

378 There is no written documentation of what should constitute "acceptable" for each test. In many tests, the "error" (difference between COBEF result and reference solution) is less than 1%, which can be considered satisfactory. In some tests, the COBEF result is significantly different from the reference solution: "error" greater than 5% was noted in

several cases. There is no written discussion about the reasons for such differences. The impact of these errors should be investigated for their influence on any use of COBEF for design of safety-related structures Assessment Finding **AF-UKEPR-CE-09**.

379 COB participated in a few validation projects organised externally comparing analysis predictions against experimental results, including CESA, MAEVA and TACIS as well as predictions of the effects of pre-stressing on the Civaux containment. The results of these predictions are satisfactory.

4.3.4.5.8 Code ASTHER HERAST

380 The scope of the evaluation of Code ASTHER and HERAST is as follows:

- Establish the full extent of the use of ASTHER and HERAST in the analysis and design process of civil structures of the UK EPR and identify those functions which will be examined in more detail.
- Understand and review the development history, development process, third party reviews, application history and any regulatory oversight/review of this pair of computer software codes.
- Review and assessment of verification, validation, qualification and Quality Assurance processes.
- Establish any limitations of ASTHER and HERAST.

381 Ref. TSC23 contains a thorough review of the code, the key points of which are repeated below

382 ASTHER and HERAST have been used by IOSIS on the following list of buildings and structures of the EPR:

- The Common Raft (ASTHER).
- The Fuel Building (both codes).
- Safeguard Auxiliary Building divisions 2 and 3 (IOSIS scope, both codes).
- Safeguard Auxiliary Building divisions 1 and 4 (CoB scope, both codes).
- The APC Shell (excluding aircraft impact analysis, ASTHER.

383 ASTHER and HERAST do not perform any computational mechanics analysis tasks. The pair of computer software codes only acts as an interface between Code ASTER and HERCULE, transferring finite element mesh, input data and finite element analysis results from one to another, more specifically:

- ASTHER transfers Code ASTER finite element mesh, input data and finite element analysis results to their counterparts in HERCULE.
- ASTHER was used to transfer results obtained from Code ASTER to HERCULE results files for the Fuel Building, the four divisions of the Safeguard Auxiliary Building, the APC shell and the common Raft.
- HERAST transfers HERCULE finite element mesh, input data but not any finite element analysis results to their counterparts in Code ASTER.
- HERAST was used to transfer the finite element mesh, beam elements, shell elements, boundary conditions and groups of masses from HERCULE to Code

ASTER for the Fuel Building and the four divisions of the Safeguard Auxiliary Building. No load cases are transferred by HERAST.

384 The two computer codes were developed in 2006 – 2007 for use with version 6 of Code ASTER. No information on the updating of them has been provided by EDF and AREVA. Revision of this pair of codes may be necessary to match the evolution of Code ASTER from version 6 to future current versions.

385 The development personnel of ASTHER and HERAST appear to have the required educational qualifications, work experiences and computer programming skills, as well as a long term employment association with IOSIS. It has been noted that the developer of these two codes have left IOSIS.

386 The development process, approval procedure and QA procedures appear consistent with those of other scientific computer software codes, for instance Code ASTER and are acceptable.

387 The test methods used to validate ASTHER and HERAST are appropriate and acceptable.

4.3.4.6 Code CASTEM (GIBI)

388 Code CASTEM GIBI was used as a front-end pre-processor for Code ASTER which performs all the static and dynamic computational mechanics analysis tasks. CASTEM was used for all the Nuclear Island buildings and structures on the cruciform common raft foundation, including the following:

- The reinforced concrete common raft structure.
- The Containment Internal Structures.
- The Inner Containment.
- The APC Shell.
- The Outer Containment.
- The Fuel Building.
- All divisions of the Safeguard Auxiliary Building.

389 Ref. TSC25 contains a thorough review of the code, the key points of which are repeated below.

390 The EDF and AREVA response to TQ-EPR-856 confirmed that only the pre-processor of CASTEM called GIBI has been used. Hence, CASTEM/GIBI has been used as the pre-processor for Code ASTER only. EDF and AREVA further confirmed that no computational tasks has been performed using CASTEM.

391 The review of CASTEM (GIBI) has confirmed the following:

- The pre-processor of CASTEM, GIBI, was used as the front-end to Code ASTER for constructing the Code ASTER global model of the Nuclear Island structures on the common raft.
- No analysis capabilities of CASTEM have been used on the Flamanville 3 EPR civil structural works.

- CASTEM/GIBI has been used for performing analysis tasks as well as for creating finite element models of civil structures of both nuclear power and non-nuclear power projects.
- CASTEM has been developed and maintained by CEA continuously since 1983 with annual releases.
- CASTEM has a wide user base in education, research and industrial sectors. Notably it is used in the nuclear power industry by all departments of CEA and by EDF, AREVA NP, AREVA TA, AREVA NC, and IRSN.
- A wide range of capabilities of CASTEM has been validated and benchmarked against analytical solutions and NAFEMS published benchmark tests.

392 In summary, CASTEM/GIBI appear broadly suitable for application in the civil structural design and analysis of nuclear power projects in the UK.

4.3.4.6.1 Code HER 2 COB

393 HER2COB does not perform any computational mechanics analysis tasks. It is an interface software tool between HERCULE used by IOSIS and COBEF used by COB. HER2COB is capable of reading in a HERCULE results data base file (.res file) in ASCII text format and translate to three COBEF results database files.

394 Ref. TSC25 contains a thorough review of the code, the key points of which are repeated below.

395 Since HER2COB was developed for two very specific structures, its capabilities are have been developed to do just enough for the job and therefore are limited to the size and the spectrum of finite element results in the SAB 1 and 4 models.

396 The EDF and AREVA response to TQ-EPR-1393 stated that no Quality Assurance procedure was implemented in the development process of HER2COB. The reason given by EDF and AREVA for the lack of a QA procedure in the development of HER2COB was that it is a one-off computer program.

397 The EDF and AREVA written response to TQ-EPR-1396 stated that HER2COB is a computer software tool transferring results from one format to another. Consequently, EDF and AREVA consider that HER2COB cannot be considered as a scientific application which they define as computer codes modelling physics equations and/or performing complex mathematical computations. Therefore, EDF and AREVA consider HER2COB is outside the scope of the French standard AFNOR Z61-02. EDF and AREVA further stated in their response to TQ-EPR-1396 that since HER2COB is not a scientific application, a formal verification and qualification were not recorded.

398 The EDF and AREVA response to TQ-EPR-1394 confirmed that no documentation is available on HER2COB, although EDF and AREVA stated that the ASCII text output of HER2COB was checked manually during the transfer of results. This approach is not considered sufficient to verify and to validate HER2COB, since the HER2COB output contained in the two binary output files cannot be checked manually.

399 It is an assessment finding that the licensee shall demonstrate that adequate verification and validation of any results from the use of HER2COB is undertaken (Assessment Finding **AF-UKEPR-CE-10**).

400 A review of HER2COB has reached the following conclusions:

- HER2COB was used by COB for SAB 1 and 4 of the Flamanville 3 EPR project in France in 2006.
- It does not perform any analysis or calculation tasks but merely converts the global Nuclear Island structural analysis results from HERCULE format to COBEF format.
- Future use of HER2COB is not expected.
- The lack of an implemented QA procedure in the development of HER2COB is considered a shortfall.
- The lack of recorded Verification and Validation of HER2COB and the fact that only manual checking was performed which cannot check the binary output of HER2COB is considered a shortfall.

4.3.4.6.2 COBEF to HERCULE Interface

401 The COBEF to HERCULE conversion computer software was used on the Safeguard Auxiliary Building divisions 1 and 4 only. These two buildings are in the analysis and design scope of COB which uses COBEF to perform finite element analysis tasks other than those that must be performed using the global Code ASTER nuclear island model.

402 Ref. TSC25 contains a thorough review of the code, the key points of which are repeated below.

403 The COBEF to HERCULE conversion interface was developed as a Microsoft Excel workbook containing several Excel spreadsheets. It reads in the ASCII text files of a COBEF finite element model and converts them to their HERCULE finite element model counterparts.

404 This computer interface tool uses a few simple Excel functions only. No macros are used. The layout of the spreadsheets and the functions used are simple, transparent and easy to understand, to check and to verify.

405 In summary, the conversion process is seen to be transparent and suitable for the design of the SAB 1 and 4 buildings.

4.3.4.6.3 Code SIGNSOLL

406 The seismic analysis of the civil structural works of the Nuclear Island structures on the common cruciform raft was performed using Code ASTER using the multi-modal response spectrum analysis method. The combination of peak modal responses is computed employing the CQC method. Only the peak values of the various response parameters are obtained by this analysis method without any information on the signs (positive or negative) or phase/timing of the peak seismic responses.

407 Ref. TSC25 contains a thorough review of the code, the key points of which are repeated below.

408 For a shell element, Code ASTER reports the peak positive values of the 6 seismic force components – 3 membrane normal and shear forces, two out-of plane bending moments and one torsional moment. This all-positive combination of the 6 shell force components often is not the most unfavourable. The most unfavourable and conservative combination of the 6 shell seismic force components needs to be identified and combined with the seismic force components arising from other load cases. All permutations of the + and – signs of the 6 shell force components need to be considered. This method results in a

total 64 combinations for each of the three components (two horizontal and one vertical) of the earthquake ground motion input.

409 This task was performed outside Code ASTER and HERCULE by a special purpose computer software code SIGNSOLL manipulating the HERCULE results database.

410 SIGNSOLL has been used in the seismic analysis of the Nuclear Island structures on the common raft foundation. The seismic analysis results of the following structures have been processed by SIGNSOLL before load combinations are performed:

- The common raft.
- The Fuel Building.
- Safeguard Auxiliary Building divisions 1 through to 4.
- The APC Shell.
- The Outer Containment.

411 SIGNSOLL does not perform any calculations but merely generates 64 combinations of positive or negative sign for each one of the 6 shell element internal force components.

412 A review of SIGNSOLL has reached the following conclusions:

- SIGNSOLL has been developed by IOSIS to meet the needs of the Flamanville 3 EPR project and has been used for this project only. There has been only one release to date – the 2007 release.
- Its development follows the standard Quality Assurance procedures of IOSIS for computer software development. Its QA procedure and its key QA documents are acceptable.
- SIGNSOLL is considered an elementary computer software code and as such is considered by IOSIS not requiring any 3rd party reviews, nor any regulatory reviews/oversights.
- Validation test and validation documentation of SIGNSOLL are acceptable.

413 It appears that SIGNSOLL is suitable for application in the design and analysis of civil structural works of nuclear power projects.

4.3.4.6.4 COBEF to NASTRAN Convertor

414 The structural analysis and design of the Nuclear Auxiliary Building (hereafter referred to as the NAB) were carried out in two stages and by two organisations.

415 COB completes a preliminary design of the NAB, including the following scope of design and analysis work:

- Definition of methodology and hypotheses.
- Creation of a 3D global finite element analysis model using COBEF.
- Definition of load cases.
- Combination of load cases.
- Calculation of required steel reinforcement for all load combinations.
- Generation of the envelope of needed reinforcement.

- Generation of maps of reinforcement.

416 Ref. TSC25 contains a thorough review of the code, the key points of which are repeated below.

417 Nuclear Design Associates (NDA) is a joint venture company formed by Sir Robert McAlpine Ltd and Taylor Woodrow Construction Ltd. NDA was appointed by COB in 2006 as a subconsultant to perform the detailed design of the NAB, including the following scope of analysis and design work:

- Transfer the global COBEF finite element analysis model and results to NASTRAN used by NDA.
- Develop the COB global model to more detailed local analysis models using NASTRAN
- Detailed design of walls and slabs by considering local load cases, the new arrangement of walls, the new openings, the new forces applied by equipment and a more accurate distribution of moment in slabs.
- Produce formwork drawings.
- Produce reinforcement guide drawings.

418 A key aspect in the COB – NDA interface is the transfer of the COBEF global finite element analysis model to that of NASTRAN.

419 The COBEF to NASTRAN conversion computer software code is named COB_READ and is written in the FORTRAN programming language. It is capable of reading in a COBEF finite element model data file and writing out a NASTRAN equivalent.

420 A review of COB_READ has reached the following conclusions.

- COB_READ was developed by COB's subconsultant NDA for the detailed design of the NAB. It does not perform any computations but merely convert COB's NAB global finite element model in COBEF format to a NASTRAN equivalent.
- COB_READ is unlikely to be used in the future since Sir Robert McAlpine now uses ANSYS to perform finite element analysis.
- The NASTRAN finite element data entities selected to be the counterparts of those in the COBEF model are appropriate.
- Development of COB_READ followed the Quality Assurance procedures of Sir Robert McAlpine.
- The NDA calculation report is a comprehensive documentation of COB_READ, serving the purpose of its theory manual, user's manual and its verification and validation report.
- The NASTRAN model of the NAB structure created by COB_READ has been verified and benchmarked against the COBEF model comprehensively and rigorously. The consistency between the two finite element models can be considered acceptable

421 In summary, the COBEF to NASTRAN convertor is considered acceptable.

4.3.4.6.5 Summary

422 Code ASTER has been found suitable for the analysis of the EPR nuclear civil structures under static and seismic loadings.

423 Code ProMISS3D would require further justification if it was to be applied for use in the design of facilities in the UK where the ground conditions are not sufficiently hard.

424 Code Europlexus has been found suitable for the analysis of the EPR nuclear civil structures.

425 Code SYSTUS has been found suitable for the analysis of the EPR liner and penetrations.

426 The codes PRECONT and FERRAIL are considered appropriate for use in the design of the containment structure.

427 Code SOFISTIK has been found suitable for the analysis of the EPR nuclear civil structures.

428 Code HFERCOQ has been found suitable for the design of reinforced concrete sections.

429 Code HERCULE has been found suitable for the analysis of the EPR nuclear civil structures under static and seismic loadings.

430 Code COBEF has been found suitable for the analysis of the EPR nuclear civil structures under static and seismic loadings, notwithstanding Assessment Finding **AF-UKEPR-CE-09**.

431 Code ASTHER/HERAST has been found to be suitable to transfer finite element meshes between ASTER and HERCULE and vice versa.

432 Code CASTEM (GIBI) has been found suitable for the generation of Finite element meshes for input into Code ASTER.

433 Code HER 2 COB has found to be lacking adequate verification and validation.

434 The COBEF to HERCULE interface software has been found to be adequate.

435 Code SIGNSOLL has been found suitable for the manipulation of seismic forces from HERCULE.

436 The COBEF to NASTRAN interface software has been found to be adequate.

437 Code NASTRAN has been found suitable for the analysis of the EPR nuclear civil structures under static and seismic loadings.

438 Code ANSYS has been found suitable for the analysis of the EPR nuclear civil structures under static and seismic loadings.

439 The following Assessment Findings have emerged:

> ***AF-UKEPR-CE-08:*** *The licensee shall justify the use of MISS3D if used as a design tool for UK sites. Justification of its use and the completion of additional studies may be required. This shall be undertaken ahead of the placement of any Nuclear Island structural concrete to ensure that the design of these structures is done in an acceptable manner.*

> ***AF-UKEPR-CE-09:*** *The licensee shall undertake a review of the implications of the discrepancies in the COBEF qualification documents between the reference solution and the COBEF result. Before structures are constructed which have used COBEF in their design, the impact of this review should be evaluated and any necessary design changes incorporated. This shall be undertaken ahead of the placement of any Nuclear Island structural concrete in structures which have used COBEF in their design.*

> ***AF-UKEPR-CE-10:*** *The licensee shall demonstrate that adequate verification and validation of any results from the use of HER2COB is undertaken. This shall be undertaken ahead of the placement of any Nuclear Island structural concrete in structures which have used HER2COB in their design.*

4.3.5 Nuclear Island Structures

440 The common cruciform raft of the Nuclear Island supports the bulk of the safety critical civil structures including the inner containment, safeguard buildings, fuel building and the aircraft shell.

4.3.5.1 Scope

441 The APC shell is discussed in Section 4.3.7 and the inner containment is discussed in Section 4.4.7. This section discusses the Safeguards Buildings and the Fuel Building and the common raft.

4.3.5.2 Standards

442 The key SAPs identified in sections 4.3.3.2, 4.3.4.2 are all relevant to the design of the Nuclear Island structures.

4.3.5.3 Assessment

443 All the Nuclear Island structures are classified as safety class C1 and seismic class SC1. In addition, safeguards buildings 2 and 3, the fuel building and the inner containment are protected against aircraft impact.

444 The common foundation raft beneath the Nuclear Island is a complex structure, the detailed design of which is outwith the scope of GDA. This is due to the high level of site dependency on the design of the common raft, and the potentially large range of site types within the UK.

4.3.5.3.1 Common Raft

445 The common foundation raft beneath the Nuclear Island supports all the key safety-related structures with the exception of the NAB which sits on its own foundation, immediately adjacent but structurally separate from the common raft.

446 In the initial stages of GDA it became apparent that detailed design of the common raft for a range of site conditions had not been performed. Rather, a detailed design had been undertaken for the Flamanville site in terms of the loads and displacements on the raft, and the reinforcement requirements calculated. The Flamanville site is a particularly hard site and is not representative of the type of sites that are generally to be found in the UK. As a result, it was agreed with EDF and AREVA that the methodologies adopted for the analysis of the raft would be examined in GDA, but that the detailed design would be excluded. In addition, EDF and AREVA agreed to provide documentation which summarised their approach, and the likely approach to be adopted for the design of a raft foundation in the UK. Refs EA11, EA78 and EA79 provide further details.

447 These references have arrived late in the Step 4 programme and have not been examined. This has been carried forwarded as a GDA Issue **(GI-UKEPR-CE-04).**

4.3.5.3.2 Global Nuclear Island Hypothesis Note

448 Ref. EA12 has been reviewed in detail, and the following noted:

- The document is specific for Flamanville, including definition of ground conditions, climatic conditions and the structural classification detailed in Appendix 2 of Ref. EA12.
- An overall design life of 85 years is identified.
- The note covers all those structures on the Nuclear Island raft, the NAB, access tower building, effluent treatment building and the diesel building.
- Extensive references are made to French legislation and decrees and standards as well as the Falmanville 3 PSAR.
- A number of the key references have been superseded.
- The document does not provide sufficient information on how load drops will be treated in the design.
- The treatment of load combinations does not fully align with the 2010 version of ETC-C and the UK companion document requirements.
- There are no apparent requirements to consider robustness or global stability of the NI structures.
- There is no reference to the need to consider decree 93-1418 or 94-1159, the French equivalent of the CDM regulations.
- The document lacks detail in a number of areas including structural philosophy, analysis methods, interfacing with adjacent structures etc.
- The sections on the treatment of earthquakes and foundations are inconsistent with the latest methodologies in Refs EA11 and EA78.

449 Ref. EA12 has been reviewed in detail, and it is considered that rewriting is necessary to enable its use for a UK EPR. This is considered a GDA Issue and has been taken forward as part of **GI-UKEPR-CE-01**.

4.3.5.3.3 Global Nuclear Island Specification Documents

450 There are a series of documents which provide generic information on aspects of design and construction of the EPR civil structures, Refs EA13, EA14, EA15 and EA86.

General Specification

451 Ref. EA13 provides a series of standard details for use in the civil engineered structures of EPR. These include anchorages, sumps, embeddments, earthing system, survey points, doors, openings and waterproofing. For various reasons only a small aspect of this document remains in-scope.

452 The anchors as presented in Ref. EA13 have been designed against the ETC-C Revision B. The ETC-C AFCEN has different rules and the designs will need to be re-validated before deployment. (this is captured in the general assessment finding of a need to re-visit calculations undertaken by ETC-C revision B to confirm their ongoing acceptability

under the requirements of ETC-C AFCEN and the UK companion document) (Assessment Finding **AF-UKEPR-CE- 07**).

453 Section 3.5 of Ref. EA13 provides some details on double lined drains cast into the floor slab. Generally, we would try to avoid the use of cast in place floor drains, preferring the use of drains within trenches that can be inspected regularly. Where this cannot be provided, we would expect the drains and the secondary containment to be capable of being tested, and that this is embodied within the maintenance arrangements. The licensee will need to ensure that whenever cast in concrete drains with secondary containment are used, they shall be capable of being pressure tested (Inner and outer boundary) (Assessment Finding **AF-UKEPR-CE-12**). This should be completed ahead of the placement of Nuclear Island safety-related concrete, which would obscure and potential modifications to the design of the drains.

454 Section 9 of Ref. EA13 provides typical details for the earthing system, including an overview of the functionality. The system appears to be required in all concrete structures on the Nuclear Island.

455 The earthing system appears to have two key functions. Firstly to provide a conduit for lightning through the structure and secondly to provide protection against personnel in the event of an electrical fault developing. The description of the arrangements for the welding of rebars to ensure continuity of the electrical path are unclear. Given the potential for welding of reinforcement to cause localised weaknesses, there is a need for greater clarity of the extent of any welding and identification of control measures anticipated. I am therefore raising an assessment finding that greater details of the extent of welding required to create the Faraday Cage need to be provided along with a suitable justification that the use of welding will not compromise the primary function of the reinforcement. The licensee shall provide further details of the earthing and Faraday Cage system such that the impact on the reinforcement bars used can be established. Where welding of reinforcement is proposed, this should be justified (Assessment Finding **AF-UKEPR-CE-13**). This should be completed ahead of the placement of Nuclear Island safety-related concrete, which will render any modifications to the Faraday Cage impossible.

456 It is unclear if the earthing system has been designed against BS EN 62305 (Ref. OD32). This has been raised as an assessment finding in the Electrical Engineering topic area.

457 The details on movement joints and building joints in Section 14 of Ref. EA13 appear reasonable in principle. The size of the proposed gap between the Nuclear Island and the NAB however is 150mm. The practicalities of achieving a seal over a 150mm wide gap will require further consideration at a site specific level. The licensee will need to provide details of the movement joints between the Nuclear Island and adjacent structures in terms of their effectiveness, practicability and longevity (Assessment Finding **AF-UKEPR-CE-14**). This should be completed ahead of the placement of Nuclear Island safety-related concrete, as once in place there is little than can be done to modify joint arrangements.

458 Section 14.8 of Ref. EA13 provides some details on the waterproofing membrane that has been used at Flamanville. The GDA scope is restricted to acknowledging that a waterproof membrane will be provided for the base and buried sections of all structures in the UK. The approach adopted of an unbonded waterproofing will require further consideration at the site specific stage. The licensee shall provide details of the waterproof membrane for safety critical structures in terms of its effectiveness, practicability and longevity (Assessment Finding **AF-UKEPR-CE-15**). This should be

completed ahead of the placement of Nuclear Island safety-related concrete, as once in place there is nothing that can be done to modify the design.

Steel Structures

459 A review of Ref. EA14 (Design of Steel structures) has identified the following:

- The document is specifically written for the Flamanville site.
- The design life of 85 years has been stated which is appropriate.
- The note covers all those structures on the Nuclear Island raft, the NAB, access tower building, effluent treatment building and the diesel building. In addition, some of the non-Nuclear Island structures are covered.
- Extensive references are made to French legislation and decrees as well as standards.
- The guidance on seismic loading is not strictly applicable to the UK.
- There is considerable detail on load combinations and replication of aspects of the ETC-C. This may not fully align with the 2010 version of ETC-C and the UK companion documents requirements.
- There is no reference to the need to consider decree 93-1418 or 94-1159, the French equivalent of the CDM regulations.
- The document lacks detail in a number of areas including structural philosophy, analysis methods, interfacing with adjacent structures etc.

460 In summary, document Ref. EA14 requires a substantial re-write to allow its use in the UK as a specifications document. There are no class 1 or 2 steel structures which are located at ground level. It is therefore not considered a GDA Issue to rewrite this code. However, it is considered an assessment finding to develop a suitable revision to this document ahead of developing site specific design of steel structures. The licensee shall develop a hypothesis note for the design of class 1 and 2 steel structures ahead of any site specific design (Assessment Finding **AF-UKEPR-CE-16**). This should be completed ahead of the installation of the polar crane. There is no direct correlation between the crane installation and the design of the steel structures, however the timing of the design fits with the polar crane being installed.

Pool Liner Design

461 The design of the pool liners (Fuel pool, IRWST tank, Reactor compartment) is captured in ETC-C, the UK companion document, Ref. EA15 and supporting information in Ref. EA86.

462 The key features of the pool liner system proposed are

- Leak tight, decontaminable and corrosion resistant metal liner made from austenitic stainless steel without molybdenum.
- The liner is made up of metal sheets at least 4mm thick except for the bottom of the pools which should be at least 6 mm thick.
- The liner is anchored to structural concrete by a system that accommodates the installation tolerances relating to the steel liner.
- A leak detection system is provided to identify leaks which emerge from incomplete welds.

- The anchoring points used to secure the liner to the first stage of civil works must be compatible with:
  - o The leak detection and drainage system perpendicular to the welds (with a slope of 1%).
  - o The weld X-ray inspection method.
- The sealed welds should be as small as possible.
- The liner has no structural function: the anchorage for the equipment must be directly transmitted to structural concrete.

463 The design choice that EDF and AREVA have proposed is the result of extensive experience and operational feedback in this area (Ref. EA86). The guidance within ETC-C Revision B was felt to be inadequate, however EDF and AREVA have provided some further information in the UK companion document (Ref. EA74). This further information is still not seen as fully sufficient to allow a design to be undertaken. Some form of hypothesis note with appropriate guidance is required for the pool liner design at the site specific stage (Assessment Finding **AF-UKEPR-CE-17**). This should be completed ahead of the placement of Nuclear Island safety-related concrete. The design of the pool structures influences the lower levels of the Nuclear Island and it is clear that this needs to be undertaken ahead of placement of concrete in these areas.

464 The arrangements for the testing of the pools is another area where the current arrangements as outlined in the ETC-C and Ref. EA15 are insufficient. The time periods for testing are noted as 1 week in the ETC-C and 1 month in Ref. EA15. The acceptance criteria are poorly described and further development of suitable acceptance criteria will be required at the site specific stage (Assessment Finding **AF-UKEPR-CE-18**). This should be undertaken before installation of the polar crane. There is no direct correlation between the pool testing and the crane, however the timing of these two activities is broadly similar.

465 The following Assessment Findings have emerged:

AF-UKEPR-CE-11: *Not used.*

AF-UKEPR-CE-12: *The licensee will need to ensure that whenever cast in concrete drains with secondary containment are used, they shall be capable of being pressure tested (Inner and outer boundary). This should be completed ahead of the placement of Nuclear Island safety-related concrete, which would obscure and potential modifications to the design of the drains.*

AF-UKEPR-CE-13: *The licensee shall provide further details of the earthing and Faraday Cage system such that the impact on the reinforcement bars used can be established. Where welding of reinforcement is proposed, this should be justified. This should be completed ahead of the placement of Nuclear Island safety-related concrete, which will render any modifications to the Faraday Cage impossible.*

AF-UKEPR-CE-14: *The licensee will need to provide details of the movement joints between the Nuclear Island and adjacent structures in terms of their effectiveness, practicability and longevity. This should be completed ahead of the placement of Nuclear Island safety-related concrete, as once in place there is little than can be done to modify joint arrangements.*

AF-UKEPR-CE-15: *The licensee shall provide details of the waterproof membrane for safety critical structures in terms of its effectiveness, practicability and longevity.*

This should be completed ahead of the placement of Nuclear Island safety-related concrete, as once in place there is nothing that can be done to modify the design.

AF-UKEPR-CE-16: *The licensee shall develop a hypothesis note for the design of class 1 and 2 steel structures ahead of any site specific design. This should be completed ahead of the installation of the polar crane.*

AF-UKEPR-CE-17: *Some form of hypothesis note with appropriate guidance is required for the pool liner design at the site specific stage. This should be completed ahead of the placement of Nuclear Island safety-related concrete.*

AF-UKEPR-CE-18: *The acceptance criteria are poorly described and further development of suitable acceptance criteria will be required at the site specific stage. This should be undertaken before installation of the polar crane.*

4.3.5.3.4 Global Nuclear Island Analysis

466 The analysis of the Nuclear Island has been undertaken for two separate reasons. Firstly, to extract forces and moments to allow detailed design of the main structures and secondly to develop in-structure response spectra to allow the design of plant and equipment within the structures. Ref. TSC26 provides further details on the assessment.

467 The derivation of the in-structure response spectra has been done for a range of soil types from soft through to the very hard Flamanville site. Figure 5 shows the range of shear modulus for the foundation along with an indication of the range of properties found in the UK. As can be seen, the range used broadly envelopes the range expected in the UK.

468 The methodologies used for the analysis have been examined in detail, and are discussed in the following sections.

4.3.5.3.5 Global Nuclear Island Analysis: Model Assembly

469 The processes of constructing (pre-processing) the global Code ASTER model of the NI structures on the common raft and post-processing the Code ASTER analysis results are complex. Many computational mechanics software codes have been used in these processes and many interfaces exist to connect the various organisations and the various computer software codes together to facilitate the work flow. Figure 4 provides a simplified overview of the process. Ref. EA80 provides further details.

470 The process of constructing the global Code ASTER analysis model of the NI structures on the common raft is documented to some detail in Ref. EA7. Further information was provided in the EDF and AREVA response to TQ-EPR-850. The whole process of building the global Code ASTER model of the NI structures on the common raft involves several steps, interfaces and computer codes.

471 IOSIS built three HERCULE finite element models for the three buildings in their scope: the Fuel Building and SAB divisions 2 and 3. A COBEF finite element model of SAB 1 was built by COB. This model was subsequently converted to a HERCULE model through a COBEF to HERCULE interface. SAB 4 is a mirror image of SAB 1 and hence its finite element model was established by a simple mirror image operation in a finite element pre-processor.

472 The 5 models above in HERCULE were converted to a Code ASTER format such that they could be manipulated by the CASTEM pre-processor GIBI.

473 Code ASTER finite element models of the following list of structures on the common raft were constructed directly using GIBI:

- The Common Raft.
- The Inner Containment.
- The Outer Containment.
- The Containment Internal Structures of the Reactor Building.
- The APC shell

474 Finally, all the models were conjoined into a single code ASTER model.

475 The individual items of software used to perform the analysis and the interfacing software have all been examined and the outcomes are reported in Section 4.2.4.

476 The model assembly has been found to have been done in an appropriate manner.

4.3.5.3.5.1 Global Nuclear Island Analysis: Soil Structure Interaction

477 In order to gain a complete understanding of the behaviour of the Nuclear Island structure, a representation of the founding material is required. At the generic stage of assessment, this is somewhat of a challenge, given the large potential range of soil conditions which may be encountered in the UK.

478 Three basic sets of analyses have been undertaken as listed below.

- Static analysis to estimate the settlement of the NI structures at the Flamanville site.
- Dynamic Time History analysis to predict the in-structure response to earthquake loading for a range of site conditions.
- Modal response spectrum analysis to predict the in-structure forces, accelerations and displacements for the Flamanville site conditions.

Each of these is discussed further in the following paragraphs.

479 A static analysis has been undertaken incorporating a soil volume of 300 m long x 300 m wide x 300m deep around and below the NI. This was modelled by solid finite elements. This model is based on the Flamanville site and has not been considered in detail for GDA. This is an area which will need to be examined in detail at the site licensing stage. The licensee shall provide the static soil analysis methodology and results for the Class 1 and 2 civil structures (Assessment Finding **AF-UKEPR-CE-19**).

480 Figure 6 shows the evolution of the Code ASTER global NI model following the development of the EPR design. EDF and AREVA confirmed that the seismic floor response spectra were computed using the Basic Design Update model of 2005 and the seismic response analysis of the NI structures on the common raft was carried out using the Detailed Design model of 2007.

481 The NI model was refined from 1996 to 2007 as the design progressed from the basic design stage to the detailed design stage. The element size was 6 m in the Basic Design model of 1996. This was refined to an element size of 2 m in the Detailed Design model of 2007. However, this refinement was not part of the EDF and AREVA intentional calculation verification process. Rather, the model became more refined and more detailed as the design progressed and more details became available.

482 The generic approach used in the ProMISS3D and ASTER interface is described in Section 4.2.4.5.1. The validity of this approach is discussed more below.

483 The ProMISS3D to Code ASTER interface RIGI_PARASOL is critical for the seismic SSI analysis of the NI structures. The 6 impedance functions corresponding to the 6 degrees-of-freedom of a rigid raft plate calculated by ProMiss3D are lumped functions, rather than distributed to the nodes of the finite element mesh of the raft.

484 TQ-EPR-733 was therefore raised to request the mechanics principles and the detailed mathematical operations of RIGI_PARASOL. In their written response, EDF and AREVA provided the following two assumptions together with the mathematical equations for deriving the 6 elastic spring stiffness properties of a 6 degree-of-freedom spring element connected to a node of the mesh of the raft slab.

- The values of the 3 translational stiffness of a spring element connected to a node of the raft is proportional to the surface area of the finite element mesh tributary to the node under consideration
- A weighting (distribution) function is used. The value of the weighting function at a node depends on the distance from the node being considered to the centre of gravity of the raft.

485 The above two assumptions are sufficient to derive the 3 translational stiffness values of each spring element based on their corresponding lumped total translational stiffness values. For deriving the 3 rotational stiffness values of a spring element, the following procedure is used.

<u>Step 1</u>

For each one of the 3 global rotational degrees-of-freedom of the common raft as a rigid body, the contribution from the translational stiffness of the distributed spring elements is summed up and deducted from the lumped total rotational stiffness value.

<u>Step 2</u>

The remainder of each of the 3 global rotational stiffness values is then distributed to the corresponding rotational stiffness of the distributed spring elements according to the same method as that used for the 3 translational stiffnesses outlined previously.

486 The methodology outlined in the written response appears to work in terms of mathematics. A deep slice examination was therefore focused on its use in the EPR seismic analysis of the NI structures from a practical perspective. This result is correct for the overall behaviour of the structure in interaction with the soil; however such a result is not physical. This mathematical process can generate unrealistic loads in the shell elements of the raft, as mentioned in the document (Ref. EA22). This is a result of the development of negative spring stiffnesses which have no physical meaning. When the size of the raft is large, the contribution to the global rocking stiffness from the vertical spring stiffness becomes large, even exceeding the total global rocking stiffness.

487 It became clear that the actual approach adopted for the Flamanville design was not exactly as described in the theory manuals for RIGI-PARASOL. TQ-EPR-323 provided a clarification of the approach used. In simple terms, the vertical behaviour is treated separately from the translational behaviour, and two separate models are run. Decoupling the behaviour in this way can be valid providing that the rocking and vertical modes are not coupled. Appropriate checks were made on the analysis to confirm that the degree of mass correlation was below 5%.

488 Within the analysis undertaken, there is no consideration of the effects of Structure Soil Structure Interaction. For large structures in close proximity, this is seen as a shortfall. This consideration should include the dynamic and kinematic effects. The site specific studies shall ensure that due regard is taken of the effects of Structure- Soil Structure Interaction in the seismic analysis of the Class 1 and 2 structures (Assessment Finding **AF-UKEPR-CE-20**).

489 Refs EA11, EA78 and EA79 provide further details on the proposed approach for the analysis of the UK EPR. These references have arrived late in the Step 4 programme and have not been examined. This has been carried forwarded as GDA Issue **GI-UKEPR-CE-06**.

4.3.5.3.5.2 Global Nuclear Island Analysis: Solution Technique

490 There are a number of key areas of the seismic analysis which were examined in more detail.

- Elements used.
- Model Construction Interfaces.
- Mesh Sensitivity.
- Time step.
- Modes extracted (cut off frequency).
- Benchmarking.

Each of these is discussed in more detail below.

491 Examination of the Level 4 (analysis/design report) document Nuclear Island – Overall 3D Model General Description – Nodes, Elements, and Thicknesses (Ref. EA7) reveal that the global Code ASTER NI model of structures on the common raft has a total of 59,398 elements, including:

- 48,578 are quadrilateral think shell elements.
- 9,463 triangular thin shell elements.
- 1,357 beam elements.

Essentially, the NI structures on the common raft are modelled by thin shell elements. Considering these structures primarily consist of reinforced concrete walls and slabs, the use of thin shell elements is appropriate.

492 Out of the total 58,041shell elements, 16% are triangular shell elements. This is a higher than anticipated proportion of triangular elements since triangular elements are less accurate compared with quadrilateral elements. The impact locally of using triangular elements is discussed in sections 4.2.5.3.4 – 4.2.5.3.7 as appropriate.

493 There are a great many interfaces in the construction of the global nuclear island model. The broad principles for the approach adopted are laid out in (Ref. EA7). The approach adopted has been reviewed and found to be acceptable. This also needs to be reviewed in concert with the interfacing software used (ASTHERHERAST, COB2HER, HER2COB) and the overall mesh manipulation software CASTEM (GIBI). The assessment of these items of software is detailed in previous sections.

494 What has not been observed in the calculations are examples of holistic review of the process which takes information from the beginning of the modelling of a structure through the interfaces to the global NI model, and then extracts results and feeds them back to the smaller sub models of individual structures. Figure 7 provides an overview for the SAB Buildings 1 and 4.

495 There was no evidence in the initial review of the documents of a mesh sensitivity study having been carried out, despite this being a requirement of the ETC-C (1.A.3.1). EDF and AREVA provided a mesh refinement calculation verification study in their written response to TQ-EPR-729. The NI finite element model of the Basic Design Update of 2002. was refined by a factor of 2, dividing each shell element to 4 elements. Results of dynamic modal properties and floor response spectra show good agreement between the two sets of results using the 2002 Basis Design Update model and the refined model of 2010. Therefore, results presented in EDF and AREVA response to TQ-EPR-729 appear to indicate that the mesh densities of the 2005 model for computing the floor response spectra for equipment qualification and the 2007 Detailed Design model are acceptable, since the latter two models have been refined compared with the 1996 and the 2002 models, with the 2005 Basic Design Update model having 10,834 elements and the 2007 Detailed Design model having 59,400 elements.

496 It therefore appears that results obtained based on the 2005 model and the 2007 Detailed Design model can be accepted on the basis of the mesh refinement study presented in EDF and AREVA response to TQ-EPR-729, although the lack of an intended mesh refinement calculation verification is considered a concern, as it indicates a lack of adherence to the requirements laid down in the ETC-C.

497 It should be noted that the mesh density consideration above is for the seismic global response of the structures and interpretations over the suitability of the mesh to capture local behaviour should not be drawn from these conclusions. Discussions on the suitability of individual structures meshes is contained in sections 4.2.5.3.4. – 4.2.5.3.7.

498 The seismic floor response spectra calculations were carried out by the transient (response time history) analysis method in the modal basis. In dynamic time history analysis, calculation verification by the time step size refinement is critical for establishing confidence on the solution in the time domain. Assessment (Ref. EA23) reveals that no evidence exists to suggest that such time step size verification has ever been performed. Hence, TQ-EPR-730 was raised to seek clarification.

499 EDF and AREVA written response to TQ-EPR-730 states that time step size refinement to verify seismic analysis was carried out for the Basic Design models. However the verifications have not been recorded. The explicit integration algorithm was used with a time step size of 0.0005 seconds. This chosen time step size was verified by Code ASTER for the purpose of maintaining stability of the numerical scheme. A time step size refinement verification was performed by EDF and AREVA as part of their response to this TQ and their new results were presented in their written response and compared with results achieved previously using a time step size of 0.0005 seconds. The time step size chosen in their verification study was 5.0×10^{-5} seconds, being 1/10 of the previous size. Nearly identical results were obtained.

500 It is considered that the time step used is suitable for the analysis undertaken.

501 The seismic response analysis of the NI structures on the common raft was performed using the modal response spectrum analysis method for determining the seismic internal force and relative displacement demands. The number of modes used and the total effective mass captured are critical information on the modal response spectrum analysis

method for establishing confidence on the results obtained. In general, sufficient modes should be used in the response spectrum analysis such that the sum of the effective modal mass is not less than 90% of the total participating mass, which in the case of the NI structures considering seismic SSI is the total mass of all structures plus the mass of the common raft.

502 Examination of Ref. EA23 reveals that in general, modes up to a modal frequency of 14 Hz have been used and these modes capture 70% to 80% of the total mass of the NI. This is significantly short of the required 90% within ETC-C AFCEN.

503 This matter is further complicated by the fact that the NI structural system on the common raft has several structures on the same raft. Some of the structures, for instance the Inner Containment and the Containment Internal Structures, are free-standing and some other structures are partially free-standing and partially connected to the APC shell and the Outer Containment. In such a structural system, not only the total effective mass captured but also the effective mass of each individual building captures are important for establishing confidence on the seismic force and relative displacement demands of the individual structures. Under cover of letter EPR000772, EDF and AREVA have provided further justification of the mass participation in individual structures.

504 The response to letter EPR00772 has been examined, and is found to be inadequate. The response does not actually provide any detailed information on the actual levels of mass captured within individual structures. This would be a relatively simple activity to undertake, as Code ASTER has the necessary tools to allow captured mass per node to be extracted. This can then be summed over the individual structures to allow a better understanding of the dynamic behaviour that has been captured. This matter is captured within the GDA Issue **GI-UKEPR-CE-06.**

505 A corollary to the use of a frequency cut off is that due account needs to be taken of the missing mass; Section 1.A.6.2.2 of ETC-C requires rigid body residual response correction to be carried out for modal analysis. This is standard practice in seismic analysis i.e. Section 3.2.3 “Response Spectrum Method” of ASCE 4 – 98 (Ref. OD26).

506 This issue was discussed with the EDF and AREVA in a number of technical meetings It was confirmed that rigid body residual response correction was performed using Code ASTER. Letters EPR00635 and EPR00667 provided an extract of Code ASTER reference document R4.05.03 “Seismic response using spectral method”. Examination of the mathematics operations laid down in Section 4.3 “Static correction via pseudo-mode” of this Code ASTER reference document reveals that these are identical to those given in ASCE 4 – 98 although the notation used is different. Therefore, the theoretical basis for the rigid body residual response correction in Code ASTER is adequate.

507 However, the EDF and AREVA written response to TQ-EPR-1127 reported a noncompliance of code ASTER with his own reference documentation which was identified by EDF while performing the investigation to respond to TQ-EPR-1127. While performing the rigid body residual response correction operation, instead of using the spectral acceleration of the highest mode considered (at the plateau of the response spectrum curve at the cut-off frequency of 14 Hz), Code ASTER used the zero-period acceleration (ZPA) which is only a fraction of the former. This way of performing the analysis is acceptable under certain restricted circumstances, i.e. provided that there is no significant mode between the modal basis cut-off frequency and cut-off frequency of the seismic soil spectrum. This check had not been undertaken however.

508 This anomaly can have an impact on studies only when the seismic analysis:

- Is carried out by modal spectral method (Civil Works design);

- Uses a pseudo mode;
- Has a modal basis that does not include modes until the cut-off frequency of the seismic soil spectrum.

509 The effects of the anomaly mainly concern accelerations (acceleration values are underestimated), and not displacements or seismic efforts since the pseudo-mode has a negligible effect on displacement values.

510 The seismic design of the Nuclear Island structures are based on local displacements issued from the modal spectral analysis. As shown above, displacements are not affected by the anomaly and are in good agreement with transient (time history) computations performed, except for the internal structure of the Reactor Building where acceleration results are used for the pseudo-static method. Accelerations located where the pseudo-mode has a main effect (raft) are potentially underestimated. Investigations performed since the non-compliance was identified by EDF and AREVA confirm that there is no impact on the design of the internal structure of the Reactor Building. This investigation has not been reviewed as part of the GDA.

511 EDF and AREVA have agreed (TQ-EPR-1127) that for any UK project, further seismic studies will use a correct implementation of the modal spectral analysis or another appropriate method. The licensee shall ensure that any seismic analysis undertaken by Code ASTER takes account of missing mass in an appropriate manner (Assessment Finding **AF-UKEPR-CE-21**).

512 A related issue is the acceleration profile up the height of the Nuclear Island. Profiles are available from the transient time history calculation for all soil types and from the modal analysis for the Flamanville site. A comparison of the two profiles was undertaken in TQ-EPR-1127. Further work has been undertaken by EDF and AREVA and was presented in a technical meeting. The key to understanding the differences lies in the degree of rigid mass which is within models and the simplifications to the dynamic modelling of this mass through the use of “missing mass” corrections. EDF and AREVA have quoted some recent research (Ref. OD55) which suggests that care needs to be taken for large structures with a relatively high proportion of rigid mass and the use of pseudo modes in the missing mass correction. This matter will be addressed in more detail as part of the resolution of GDA Issue **GI-UKEPR-CE-06**.

513 The validation and verification of the model as a whole was questioned under TQ-EPR-728. The main response to this refers to Ref. EA81. This documents benchmarking between the 2007 Detailed Design model (constructed by IOSIS) and the 2005 Basic Design Update Model for floor response spectra computations (constructed by EDF SEPTEN). Benchmarking of the Code ASTER NI model was carried out by two independent teams using the same computer software. The following items were benchmarked between the IOSIS model and the EDF SEPTEN model:

- Mass, difference < 5%.
- Impedance functions for the Flamanville site-specific soil profile, show good agreement.
- The three fundamental modal frequencies and effective modal masses along X, Y and Z, show acceptable agreement.
- Modal damping of the three fundamental modes along X, Y and Z, show acceptable agreement.

- Acceleration profile along the elevation, some very significant differences near the base with the IOSIS peak accelerations obtained by the response spectrum analysis method being 0.2g, much lower than 0.3g obtained by EDF SEPTEN by the modal transient analysis method.
- Relative displacement profile along the elevation, some significant differences near the top.

Despite the differences observed in the acceleration and the relative displacement profiles between the two sets of results, the two models may be considered consistent. Agreement between the two sets of results may be improved had the EDF SEPTEN model been updated to be consistent with the 2007 Detailed Design. The earlier section of this report discusses the acceleration profiles in more detail.

4.3.5.3.5.3 Global Nuclear Island Analysis: Summary

514 Refs EA11, EA78 and EA79 provide further details on the proposed approach for the analysis of the UK EPR. These references have arrived late in the Step 4 programme and have not been examined. This has been carried forwarded as GDA Issue **GI-UKEPR-CE-06**.

4.3.5.3.5.4 Assessment: Safety Auxiliaries Building

515 A review of the various SAB specifications and hypothesis notes (Refs EA43, EA44, EA45 and EA46) is given below.

516 A review of Ref. EA43 (Sofinel Hypothesis note for SAB 2 and 3) has identified the following:

- The document is not specifically written for the Flamanville site, but for “any site in France” however extensive site specific data for Flamanville is provided.
- The design life of 85 years has been stated which is appropriate.
- Extensive references are made to French legislation and decrees as well as standards.
- The document acts a signpost to other hypothesis notes primarily the overall NI hypothesis note (Ref. EA12).
- There is considerable detail on load combinations and replication of aspects of the ETC-C. This may not fully align with the 2010 version of ETC-C and the UK companion documents requirements.
- There is no reference to the need to consider decree 93-1418 or 94-1159, the French equivalent of the CDM regulations.
- The document lacks detail in a number of areas including structural philosophy, analysis methods, interfacing with adjacent structures etc.

517 In summary, document Ref. EA43 requires a substantial revision for use in the UK as a specifications document.

518 A review of Ref. EA44 (Sofinel Hypothesis note for SAB 1 and 4) has identified the following:

- The document is not specifically written for the Flamanville site, but for *"any site in France"*, however extensive site specific data for Flamanville is provided.
- The design life of 85 years has been stated which is appropriate.
- Extensive references are made to French legislation and decrees as well as standards.
- The document acts a signpost to other hypothesis notes primarily the overall NI hypothesis note (Ref. EA12).
- There is considerable detail on load combinations and replication of aspects of the ETC-C. This may not fully align with the 2010 version of ETC-C and the UK companion documents requirements.
- There is no reference to the need to consider decree 93-1418 or 94-1159, the French equivalent of the CDM regulations.
- The document lacks detail in a number of areas including structural philosophy, analysis methods, interfacing with adjacent structures etc.

519 In summary, document Ref. EA44 requires a substantial revision for use in the UK as a hypothesis document.

520 A review of Ref. EA45 (IOSIS Hypothesis note for SAB 2 and 3) has identified the following:

- The document is Flamanville specific.
- A more correct design lifetime of 85 years is claimed.
- The foundation conditions are limited to those of Flamanville.
- The guidance on the construction of the finite element models for the structure are very weak without reference to other guidance.
- There is considerable detail on load combinations and replication of aspects of the ETC-C. This may not fully align with the 2010 version of ETC-C and the UK companion documents requirements.
- There is a lack of a clear structural philosophy laid out in the document.
- There is no reference to the need to consider decree 93-1418 or 94-1159, the French equivalent of the CDM regulations.

521 In summary, document Ref. EA45 requires a substantial re-write to allow its use in the UK as a detailed structure specific hypothesis document.

522 A review of Ref. EA46 (COB Hypothesis note for SAB 1,4) has identified the following

- The document is Flamanville specific.
- A more correct design lifetime of 85 years is claimed.
- The foundation conditions are limited to those of Flamanville.
- The use of an equivalent static load method for seismic cases is suggested, which is out with the requirements of ETC-C.
- The guidance on the construction of the finite element models for the structure are very weak without reference to other guidance.

- There is considerable detail on load combinations and replication of aspects of the ETC-C. This may not fully align with the 2010 version of ETC-C and the UK companion documents requirements.
- The treatment of APC scenarios is unclear.
- There is a lack of a clear structural philosophy laid out in the document.
- There is no reference to the need to consider decree 93-1418 or 94-1159, the French equivalent of the CDM regulations.

523 In summary, document Ref. EA46 requires substantial revision for use in the UK as a detailed structure specific hypothesis document.

524 Taking account of all the above comments, it is considered that the suite of documents examined Refs EA43 to EA46 do not have sufficient information to consider them adequate for use in the UK as hypothesis notes for the design of the SAB.

525 The detailed design of the SAB has not been examined within GDA. The design for Flamanville 3 has been a steady evolution as plant requirements have developed.

526 I have reviewed the basic layout of the structures and am confident that they can be shown to work. It is therefore an assessment finding that the licensee shall undertake detailed structural design of the SAB building and provide suitable justifications for the structural forms and reinforcement (Assessment Finding **AF-UKEPR-CE-22**).

4.3.5.3.5.5 Findings Fuel Building

527 A review of the various Fuel Building specifications and hypothesis notes (Refs EA38, EA39 and EA40) is given below.

528 A review of Ref. EA38 (Fuel Building Specification by EDF CNEN) has identified the following:

- The document is not specifically written for the Flamanville 3 site, in fact the document is non committal over its application.
- Some sections of the document are incomplete.
- Extensive references are made to French legislation and decrees as well as standards.
- There is no reference to the need to consider decree 93-1418 or 94-1159, the French equivalent of the CDM regulations.
- The document lacks detail in a number of areas including structural philosophy, analysis methods, interfacing with adjacent structures etc.

529 In summary, document Ref. EA38 would require a substantial revision to enable its use in the UK as a specifications document.

530 A review of Ref. EA39 (Fuel Building Hypothesis note by Sofinel) has identified the following:

- The document is specifically written for the Flamanville site.
- The design life of 85 years has been stated, with an operational life of 60 years. This is not appropriate for the fuel building.
- The document not sufficiently detailed.

- There is no reference to the need to consider decree 93-1418 or 94-1159, the French equivalent of the CDM regulations.
- The document does not cover a number of areas including structural philosophy, analysis methods, interfacing with adjacent structures etc

531 In summary, document Ref. EA39 would require a substantial revision for to enable its use in the UK as a hypothesis document.

532 A review of Ref. EA40 (Fuel Building Hypothesis note by IOSIS) has identified the following:

- The document is Flamanville 3 specific.
- The design life of 85 years has been stated, with an operational life of 60 years. This is not appropriate for the fuel building.
- The foundation conditions are limited to those of Flamanville 3.
- The guidance on the construction of the finite element models for the structure is limited and needs to be supplemented by other guidance.
- There is considerable detail on load combinations and replication of aspects of the ETC-C which may not fully align with the 2010 version of ETC-C and the UK companion documents requirements.
- There is a lack of a clear structural philosophy laid out in the document.
- There is no reference to the need to consider decree 93-1418 or 94-1159, the French equivalent of the CDM regulations.

533 In summary, document Ref. EA40 would require a substantial revision to enable its use in the UK as a hypothesis document.

534 The detailed design of the fuel building has been examined in detail (Ref. TSC33). This has focussed on those elements seen as the most critical in terms of design and safety claims.

535 Figure 1 below shows the document structure for the fuel building design.

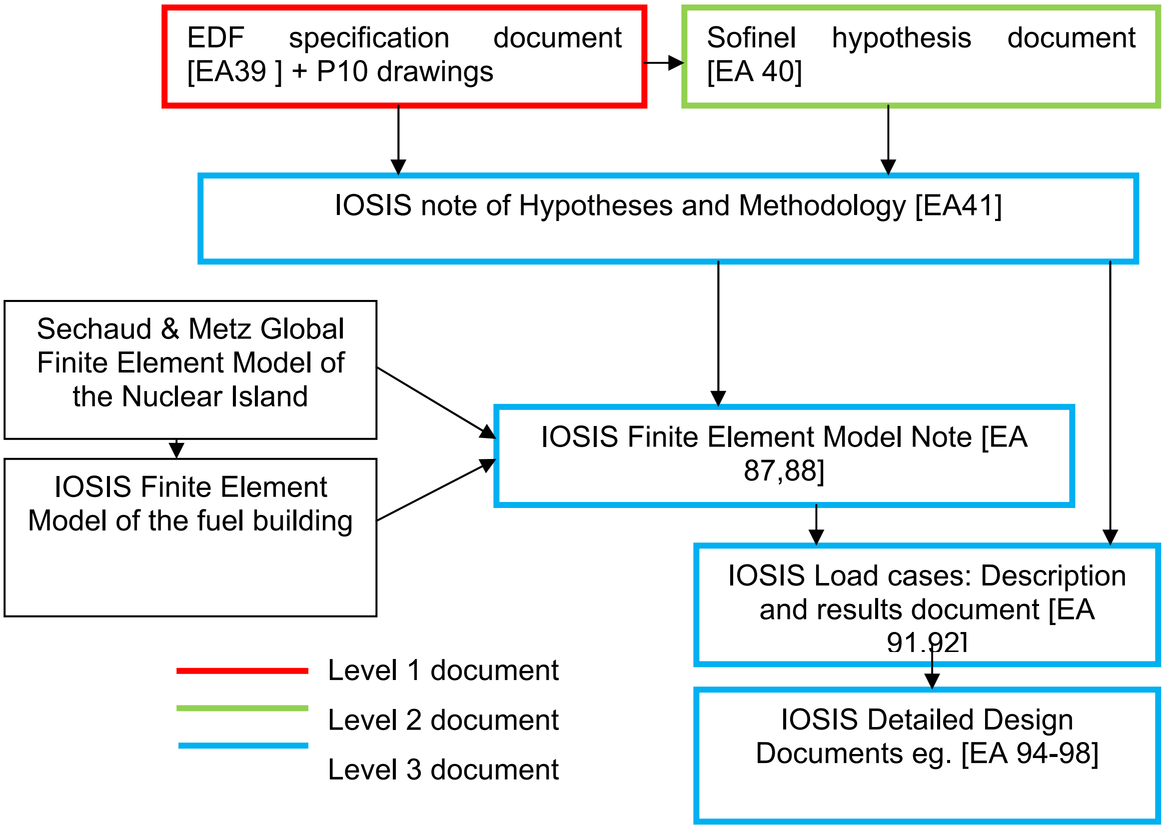

Figure 1: Document Structure for the Fuel Building Design

536 A more complete overview of the design process can be seen in Figure 8.

537 As part of the detailed review of the Fuel Building, Refs EA87 to EA98 have been examined.

538 The following aspects of the design were examined in detail:

- A tank support slab at level +3.7m.
- A transfer wall at level 0.0m to +3.7m under the spent fuel pool.
- A slab at level 0.0m which is supported on three edges and modelled with one element across its width.
- A slab at level -3.4m which was modelled using a finite element mesh of triangular elements.
- The fuel pool design.

539 Some common themes have emerged as part of the deep slice review. It is noted that additional reinforcement is calculated in the process “Additional localized calculation” It is understood that this reinforcement is added to reinforcement calculated earlier in the design process. This is a potentially unconservative approach at ULS as it does not account for the reduced lever arm associated with an enlarged stress block. This has not

been addressed in the calculations. It is therefore an assessment finding that the licensee shall ensure that wherever reinforcement areas have been added directly to produce a composite quantity, complimentary checks are undertaken to ensure that this approach is conservative (Assessment Finding **AF-UKEPR-CE-23**).

540 In Ref. EA40, some load-cases listed in ETC-C do not appear to have been examined, and the rationale for their exclusion has not been detailed. These are: 1b, 1d (construction), 3a (Normal operating +climatic), 5 (groundwater), 14a,b (flooding). It is therefore an assessment finding that the licensee shall ensure that wherever ETC-C load-cases are dismissed from the design process that a rationale for this is provided (Assessment Finding **AF-UKEPR-CE-24**).

541 The finite element mesh used to carry out the structural analysis of the fuel building is coarse (consisting of shell elements which are approximately 1m x 1m) and its appropriateness does not appear to have been verified. A number of TQs have been raised on this issue (TQ-EPR-848, TQ-EPR-859 and TQ-EPR-1031). It is noted that ETC-C does require that a mesh sensitivity analysis be carried out to ensure that refining the mesh does not lead to different results. Based upon the response to the TQs referred to above, this does not appear to have happened as part of the design process. At the site specific stage the licensee will need to ensure that for the fuel building, the simplifications in the mesh used for the structure and the applications of loads are systematically reviewed and justified (Assessment Finding **AF-UKEPR-CE-25**).

542 The output from the finite element mesh is presented directly as reinforcement areas. Stress resultant plots (e.g. moment per unit length) are not presented in any of the detailed design documents (only reinforcement intent). This makes it difficult to see equilibrium discrepancies and raises a concern that the finite element output may not have been assessed for reasonableness in terms of equilibrium. The documents reviewed do not contain a demonstration of output verification.

543 The suspended slab at the 3.7m level supports a series of ASG tanks. These are large steel tanks, 10m in height and 2.5m in diameter. It was not possible to apply the loads from the tank direct to the finite element mesh of the slab due to coarseness of the mesh. The derivation of the loads was a complex process, as a result of iterations on the support arrangements for the tanks. Initially they were free standing and eventually (to accommodate the seismic loads) they have been provided with an additional lateral support in the upper section of the tank.

544 The information in terms of loads to apply from the tank to the slab has been provided to the design team from the tank suppliers. There was limited evidence of an independent review of the information provided by the tank suppliers. For the eventual design, the degree of interaction between the structure and the tank is increased due to there being 2 points of contact, which have non-coincident motions. This degree of complexity requires more careful consideration than was evident from the documents sampled.

545 It is an assessment finding that manufacturer supplied information does not appear to have been reviewed from an intelligent customer perspective. For detailed site design, this will need to be undertaken for all manufacturer supplied information. The licensee shall review all supplier provided data used in the design of civil structures and confirm their acceptance of it as suitable for that purpose (Assessment Finding **AF-UKEPR-CE-26**).

546 The coarseness of the mesh has resulted in the tank loads being applied in positions which are not fully representative of where the tank actually is located. The implications of this for the design do not appear to have been fully considered. This is an assessment

finding. Where simplifications over the application of significant plant masses have been made, the implications for the design of supporting elements needs to be fully justified (see Assessment Finding **AF-UKEPR-CE-25**).

547 The transfer wall at level 0.0m to level +3.7 m under the spent fuel pool has been examined (Ref. EA98). The finite element model of this part of the structure does not properly represent the layout of openings in the wall. In addition, concerns were raised over the mesh density in this area. The response to TQ-ERPR-849 provided some reassurance that the mesh is reasonable for design of the wall if the apertures were not present.

548 It is recognised that the layout of the structure has evolved during the design process; this is a normal process. What is less clear from the documentation however is the degree of reconciliation of the final design documentation against the final layout. TQ-EPR-862 requested some information on this aspect for the SAB, NAB and Fuel building. A detailed response was provided for the NAB. Additionally it was stated “For future EPR projects, the Finite Element model will be updated to take into account the final geometrical changes applied to the buildings (i.e. based on final layout detailed design) and as a result the model will be more accurate/detailed”. This is not realistically a commitment which EDF and AREVA can undertake, however it will be required by a licensee. The licensee shall update the finite element models of Class 1 and 2 structures will to take into account the final geometrical changes applied to the buildings (i.e. based on final layout detailed design) (Assessment Finding **AF-UKEPR-CE-27**).

549 A review of the design of a triangular cantilever slab at the -3.4m level was undertaken. A review of Ref. EA96 and the response to TQ-EPR 848 considered the design to be adequate.

550 The cantilever slab at level 0.0 m was selected for evaluation of the use of finite element because it was only modelled with one element in the span of the cantilever. A sensitivity study on the results from out of plane behaviour was undertaken. The following is noted about these results:

- The results cannot be reconciled with equilibrium. This is partly because the mesh coarseness makes it impossible to associate a particular loaded region with the three output points.
- From equilibrium considerations alone the expectations for transverse shear (TZX) in the slab was approximately 1kN per m of slab. But the peak shear reported is 5.38kN per metre.
- From equilibrium considerations alone the expectations for moment M_{xx} in the slab was less than 1kN per m of slab. But the peak value of M_{xx} reported is 2.33kNm per metre.

551 It is noted that the loads predicted from the analysis are considerably larger than those expected from equilibrium of the slab alone. The explanation provided for this was that the slab loads are governed by global effects from the building as a whole. This suggests that, in the case of this slab, the consequences of the mesh coarseness (on its ability to predict local bending and shear) are small due to the fact that the reinforcement demand upon the slab is much more significantly influenced by global building behaviour.

552 The fuel pool structure has been examined from a fluid containment perspective. The structure as designed relies on a stainless steel liner to act as a primary barrier against leakage of pool water with a series of leak channels installed behind the site made welds

which aim to collect any leakage through the welds and direct it towards a collection point. The arrangement of collection points is such that the broad zone in which the leakage is occurring can be identified for further investigation. It is worthwhile noting that a leak through any part of the structure other than the welds may not necessarily be captured by the leak detection system.

553 Any leakage which does not pass into the leak detection channels will either pass through the concrete or down the liner to concrete interface. If the leakage were to pass through cracks in the concrete, there is a reasonable chance that this would be identified as white boric acid crystals tend to appear on the outer face of leak sites as the water is evaporated away. Continued leakage would be a concern for a number of reasons, including spread of contamination and degradation of reinforcement in and around the through thickness cracks. The limitation of leakage through the pool walls by control of cracking is seen as a sensible and necessary precaution.

554 The concrete portion of the pool structure is designed for exposure class XS3 on the outer face and XC1 on the inner face. This does not give any surety over the control of through thickness cracking and subsequent leakage. It is considered that it is reasonably practicable to ensure the pool structure to be water resistant against the requirements of BS-EN 1992 part 3 (Tightness class 1). The design of unlined reinforced concrete structures to perform in this way is common practice. This should be confirmed for Serviceability limit states SLS.c, SLS.f and static equilibrium states ELE. Future licensees shall confirm that the concrete portion of all steel lined concrete pools which have a permanent and potentially contaminated fluid shall be confirmed as adequate against the requirements of BS-EN 1992 part 3 (Tightness class 1) (Assessment Finding **AF-UKEPR-CE-28**).

4.3.5.3.6 Assessment: Nuclear Auxiliaries Building

555 The NAB has been analysed as a separate structure, using broadly the same approach as the main Nuclear Island. The seismic time history analysis has been undertaken for the full range of soil conditions within Code ASTER. It should be noted that the same limitations on the use of ProMISS 3D for the NI also apply to the NAB. The acceleration and displacement profiles up the structure have been calculated (Ref. EA84).

556 The acceleration profiles are then used to apply a pseudo-static load to a COBEF model of the structure to extract seismic loads for the key structural elements (Ref. EA53 and 85). This approach is not permitted by the ETC-C without detailed justification. TQ-EPR 861 requested further by justification of the approach. Three key reasons were given, consistency with the Flamanville 3 Preliminary report of Public Safety, the USNRC standard review plan (Ref. OD27) and IAEA TECDOC 1250 (Ref. OD28). Each of these is discussed below.

557 The use of the quasi-static structure analysis method is stated in Section 2.1.3.4 of Chapter 3.3 of the Preliminary Report of Public Safety as “*The accelerations of rigid body of the floor response spectrum, for the soil conditions of the site, are used for detailed quasi-static structural analysis of buildings*”. For the UK, this is irrelevant, and not considered further.

558 In the Standard review plan, the option exists for the transfer of dynamic response data into design by whatever technique and that “*This is reviewed for technical adequacy on a case-by-case basis*”. EDF and AREVA have not provided any generic justification or structure specific information on the adequacy of this approach.

559 The TECDOC is essentially a collection of papers rather than a comprehensive standard for undertaking analysis. It is worth noting however that the paper quoted makes the point that methods used should be "generally accepted".

560 In summary, in order to justify this method, EDF and AREVA is expected to provide a claim that the seismic analysis method used by COB for the NAB is conservative compared with the dynamic analysis method used by EDF in the NAB floor response spectra computations. This claim must then be substantiated by quantitative comparisons between the interstorey shear force, interstorey torsional moment, and interstorey overturning moment of all storeys including the 1st storey in which case comparison is made between the base shear force, base torsional moment, and base overturning moment, for both the principle axes of the NAB. The claim is acceptable if such comparisons show that COB's simplified equivalent static force method gives higher seismic loads. The licensee shall demonstrate the suitability of the equivalent lateral load method for the application of seismic loads to Seismic Class 1 and 2 structures if this approach is used (Assessment Finding **AF-UKEPR-CE-29**).

561 Analysis of the NAB against overturning and sliding does not appear to have been undertaken. I have not examined the behaviour of the NAB under gravity loads. The licensee shall demonstrate the stability of the NAB in terms of sliding and overturning under seismic loading (Assessment Finding **AF-UKEPR-CE-30**).

562 The close proximity of the NAB to the Nuclear Island foundation and the large difference in mass means that there is a strong potential for Structure- Soil Structure Interaction to occur. This has not been considered in the generic design, however it will need to be examined at the site specific stage. This has previously been raised as Assessment Finding **AF-UKEPR-CE-20**.

563 One other consideration given the close proximity of the NI and the NAB is the possibility of pounding between the two structures in a seismic event. TQ-EPR-860 was raised requesting details of the joint arrangement. In addition, since this joint extends below ground level, there is a need too ensure that the gap provided cannot get filled by material (such as soil backfill), thus making it ineffective. The response to TQ-EPR-860 provided some sketches and pointed to further information in Ref. EA13. Ref. EA13 suggests that a gap between the physical structures of 150mm will be provided, with a waterbar across the joint to provide a secondary waterproof barrier. A gap of this size is considered to be adequate, however the details of the particular materials to be used will need to be examined in more detail on a site specific basis. This will need to take account of local ground conditions for longevity of the seals, water pressure, local design modifications and the interface with the global waterproofing system employed on the site in question. This is an assessment finding, previously raised in Section 4.3.5.3.3 (Assessment Findings **AF-UKEPR-CE-14** and **AF-UKEPR-CE-15**).

564 A review of the various NAB specifications and hypothesis notes (Refs EA51, EA52 and EA53) is given below.

565 A review of Ref. EA51 (NAB Specifications by CNEN) has identified the following:

- There are frequent references to the RPS not the PCSR.
- The design life is set at 60 years- this may not be sufficient if there is a period of post shutdown decommissioning required.
- The report does not conatin design guidance on the the treatment of gaps between the NAB and SAB or Fuel Building.
- There is insufficient information on future monitoring of foundation movements.

- Statements over the need to "*reduce cracking as much as possible*" do not provide sufficient guidance to a designer.
- The bulk of the standards referenced are French national standards.
- References are made to certification bodies for materials which are French in origin.
- References to standards and guidance are not sufficiently precise.
- The option for using projecting bars (bent down bars) in openings is allowed. - This is not a practice which is generally permitted in the UK for Nuclear structures.
- There are a large number of references to Règles Fondamentales de Sûreté (RFS) documents for derivation of loads. These would require further justification for application in the UK.
- There is reference to a need to satisfy French Decree 94-1159, which is the equivalent of the CDM regulations.

566 In summary, Ref. EA51 would require a substantial revision to enable its use as a specifications document for the UK EPR.

567 A review of Ref. EA52 (NAB Hypothesis note by Sofinel) has identified the following:

- The document is not specifically written for the Flamanville site, but for *"any site in France".*
- The design life of 85 years has been stated which is appropriate.
- Extensive references are made to French legislation and decrees as well as standards.
- The document acts a signpost to other hypothesis notes primarily the overall NI hypothesis note (Ref. EA13).
- The document states that long term settlement does not need to be considered, which is seen as inappropriate.
- Reference is made to the use of bent down bars.
- There is no reference to the need to consider decree 93-1418 or 94-1159, the French equivalent of the CDM regulations.
- The document lacks detail in a number of areas including structural philosophy, analysis methods, treatment of foundations, interfacing with technical galleries etc.
- There is insufficient discussion of the global stability/ sliding of the NAB building and the required acceptance criteria.
- There is no detailed discussion on the need for some floor elements to essentially be leak-tight. - Ref. EA51 suggests that some areas require this to prevent seepage of contaminated effluent into the structure.
- There is considerable detail on load combinations and replication of aspects of the ETC-C which may not fully align with the 2010 version of ETC-C and the UK companion documents requirements.

568 In summary, Ref. EA52 would require a substantial re-write to enable its use in the UK as a specifications document.

569 A review of Ref. EA53 (NAB Hypothesis note by COB) has identified the following:

- The document is Flamanville specific.
- The document refers to a Design requirements document ECEIG060786 which has not been provided in the scope of GDA.
- A more correct design lifetime of 85 years is claimed.
- The foundation conditions are limited to those of Flamanville.
- The use of an equivalent static load method for seismic cases is suggested, which appears outwith the requirements of ETC-C.
- The guidance on the construction of the finite element models for the structure is limited and would need to be supplemented by other guidance.
- There is a lack of a clear structural philosophy laid out in the document.
- There is no reference to the need to consider decree 93-1418 or 94-1159, the French equivalent of the CDM regulations.
- There is considerable detail on load combinations and replication of aspects of the ETC-C which may not fully align with the 2010 version of ETC-C and the UK companion documents requirements.

570 In summary, document EA53 would require a substantial re-write to enable its use in the UK as a detailed structure specific hypothesis document.

571 Taking account of all the above comments, it is considered that the suite of documents examined EA51, 52 and 53 do not have sufficient information to consider them adequate for use in the UK as hypothesis notes for the design of the NAB. This is part of GDA Issue **GI-UKEPR-CE-01**.

572 The detailed design of the NAB has not been examined within GDA. The design for Flamanville 3 has been a steady evolution as plant requirements have developed. I have reviewed the basic layout of the structures and am confident that they can be shown to work. The licensee will need to undertake detailed structural design of the NAB building and provide suitable justifications for the structural forms and reinforcement (Assessment Finding **AF-UKEPR-CE-31**).

4.3.5.4 Summary

573 The design classification of the Nuclear Island structures is seen to be appropriate given their safety functions.

574 The specification, hypothesis and methodology documents are considered to require significant revision to enable them to be useable in the design of the UK EPR. This is GDA Issue **GI-UKEPR-CE-01**.

575 The following Assessment Findings have emerged:

> ***AF-UKEPR-CE-19:*** *The licensee shall provide the static soil analysis methodology and results for the Class 1 and 2 civil structures. This should be completed ahead of the placement of first structural concrete, as the findings will be required to confirm the design as acceptable.*

> ***AF-UKEPR-CE-20:*** *The licensee shall ensure that due regard is taken of the effects of Structure- Soil Structure Interaction in the seismic analysis of the Class 1 and 2 structures. This should be completed ahead of the placement of first*

structural concrete, as the analysis will be required to confirm the design as acceptable.

AF-UKEPR-CE-21: *The licensee shall ensure that any seismic analysis undertaken by Code ASTER takes account of missing mass in an appropriate manner. This should be completed ahead of the placement of first structural concrete. This should be completed ahead of the placement of first structural concrete, as the seismic analysis will be required to confirm the design as acceptable.*

AF-UKEPR-CE-22: *The licensee shall undertake detailed structural design of the SAB building and provide suitable justifications for the structural forms and reinforcement. This should be completed ahead of the placement of first Nuclear Island safety-related structural concrete, as the design is required to allow the construction drawings to be finalised.*

AF-UKEPR-CE-23: *The licensee shall ensure that wherever reinforcement areas have been added directly to produce a composite quantity, complimentary checks are undertaken to ensure that this approach is conservative. This should be completed ahead of the placement of Nuclear Island safety-related concrete, since the outcome of this check may affect the design of the Nuclear Island.*

AF-UKEPR-CE-24: *The licensee shall ensure that wherever ETC-C loadcases are dismissed from the design process that a rationale for this is provided. This should be completed ahead of the placement of Nuclear Island safety-related concrete, since the outcome of this check may affect the design of the Nuclear Island.*

AF-UKEPR-CE-25: *The licensee shall ensure that for the fuel building, the simplifications in the mesh used for the structure and the applications of loads are systematically reviewed and justified. This should be completed ahead of the placement of Nuclear Island safety-related concrete, since the outcome of this check may affect the design of the Nuclear Island.*

AF-UKEPR-CE-26: *The licensee shall review all supplier provided data used in the design of civil structures and confirm their acceptance of it as suitable for that purpose. This should be completed ahead of the placement of Nuclear Island safety-related concrete, since the outcome of this check may affect the design of the Nuclear Island.*

AF-UKEPR-CE-27: *The licensee shall update the finite element models of Class 1 and 2 structures will to take into account the final geometrical changes applied to the buildings (i.e. based on final layout detailed design. This should be completed ahead of the placement of Nuclear Island safety-related concrete, since the outcome of this check may affect the design of the Nuclear Island.*

AF-UKEPR-CE-28: *The licensee shall confirm that the concrete portion of all steel lined concrete pools which have a permanent and potentially contaminated fluid shall be confirmed as adequate against the requirements of BS-EN 1992 part 3 (Tightness class 1). This should be completed ahead of the placement of Nuclear Island safety-related concrete, since the outcome of this check may affect the design of the Nuclear Island.*

AF-UKEPR-CE-29: *The licensee shall demonstrate the suitability of the equivalent lateral load method for the application of seismic loads to Seismic Class 1 and 2 structures if this approach is used. This should be completed ahead of the placement of Nuclear Island safety-related concrete, since the outcome of this check may affect the design of the Nuclear Island.*

> ***AF-UKEPR-CE-30:*** *The licensee shall demonstrate the stability of the NAB in terms of sliding and overturning under seismic loading. This should be completed ahead of the placement of Nuclear Island safety-related concrete, since the outcome of this check may affect the design of the Nuclear Island.*
>
> ***AF-UKEPR-CE-31:*** *The licensee shall undertake detailed structural design of the NAB building and provide suitable justifications for the structural forms and reinforcement. This should be completed ahead of the placement of Nuclear Island safety-related concrete, since the outcome of this check may affect the design of the Nuclear Island.*

4.3.6 Containment Structure

4.3.6.1 Scope

576 Of particular importance are the containment structures which form the most safety critical civil structures on the facility. Whilst a large number of individual components of the assessment will be examined in the other phases of this work, it is felt necessary to bring those aspects which relate to the containment together into a coordinated response. In addition, there are particular demands and requirements placed on the containment which need to be given a more considered review. In particular this includes:

- Interface between mechanical and civil structural components, i.e. penetrations, access doors, embedments etc.
- Development of LOCA loading scenarios.
- Use of material models.
- Thermal and pressure transient development and idealisation into the structural modelling.
- Development of lifetime behaviour models, creep, shrinkage, corrosion.
- Scale Model Testing.
- Leak Testing of Penetrations.
- Integrated Leak Testing (Over pressure).

577 The key Steps during Step 3 and 4 are as follows:

- Identify design codes used for containment design.
- Identify status of codes, development state, and previous regulatory engagement.
- Review application of codes, deviation, development, and interfaces between different standards.
- Review Analysis procedures undertaken.
- Review links between design basis, loading scenarios and claimed reliability
- Review MITS against best practice.

578 The scope covered in Step 4 relates primarily to the civil structure and does not include consideration of mitigation arrangements such as the containment heat removal system.

4.3.6.2 Standards

579 The key SAPs which are applicable are as indicated in Section 4.3.3.2, 4.3.4.2 and additionally as follows.

Engineering principles: civil engineering: in-service inspection and testing	**Proof pressure tests**	**ECE.21**
Pre-stressed concrete pressure vessels and containment structures should be subjected to a proof pressure test, which may be repeated during the life of the facility.		

Engineering principles: containment and ventilation: containment design	**Prevention of leakage**	**ECV.1**
Radioactive substances should be contained and the generation of radioactive waste through the spread of contamination by leakage should be prevented.		

Engineering principles: containment and ventilation: containment design	**Minimisation of releases**	**ECV.2**
Nuclear containment and associated systems should be designed to minimise radioactive releases to the environment in normal operation, fault and accident conditions.		

Where appropriate, containment design should:

a) Define the containment boundaries with means of isolating the boundary;

b) Establish a set of design safety limits for the containment systems and for individual structures and components within each system;

c) Define the requirements for the performance of the containment in the event of a severe accident as a result of internal or external hazards, including its structural integrity and stability;

d) include provision for making the facility safe following any incident involving the release of radioactive substances within or from a containment, including equipment to allow decontamination and post-incident re-entry to be safely carried out;

e) Minimise the size and number of service penetrations in the containment boundary, which should be adequately sealed to reduce the possibility of nuclear matter escaping from containment via routes installed for other purposes;

f) Avoid the use of ducts that need to be sealed by isolating valves under accident conditions. Where isolating valves and devices are provided for the isolation of containment penetrations, their performance should be consistent with the required containment duties and should not prejudice adequate containment performance;

g) Provide discharge routes, including pressure relief systems, with treatment system(s) to minimise radioactive releases to acceptable levels. There should be appropriate treatment or containment of the fluid or the radioactive material contained within it, before or after its released from the system;

h) Allow the removal and reinstatement of shielding;

i) define the performance requirements of containment systems to support maintenance activities;

j) Demonstrate that the loss of electrical supplies, air supplies and other services does not lead to a loss of containment nor the delivery of its safety function;

k) Demonstrate the control methods and timescales for re-establishing the containment conditions where access to the containment is temporarily open (e.g. during maintenance work);

l) Incorporate measures to minimise the likelihood of unplanned criticality wherever significant amount of fissile materials may be present.

The specialised and particular nature of the containment design with grouted in place pre-stressing tendons has necessitated the development of some background information on grouted in place tendons. Gifford have produced a review of the historical performance of grouted in place tendons and an overview of what is seen as current best practice in terms of installation and monitoring (Ref. TSC3). This will also be used to inform my assessment of the acceptability of the design, construction and maintenance regimes for the pre-stressing.

4.3.6.3 Description

580 The containment for the EPR reactor is a double-walled structure founded off a reinforced concrete foundation raft. The inner containment wall is constructed using pre-stressed reinforced concrete, with a steel liner plate covering its internal surface, walls, dome and support slab. This continuous membrane provides a leak-tight surface. The outer containment wall is constructed using reinforced concrete. It ensures protection against external hazards such as aircraft crash and explosion pressure waves. The containments are separated by a 1.80 m wide annulus between the inner and outer structures. The annulus is maintained at sub-atmospheric pressure to collect any leakage through the inner containment. Any leakage is filtered, before being vented to the environment.

581 The pre-stressed reinforced concrete inner containment is comprised, from bottom to top, of a:

- Cylindrical gusset.
- Truncated section.
- Cylindrical section called the ‘inner containment skirt’.
- Torispherical dome connected to the skirt by a ring.

582 It includes:

- On its internal side, a steel leak-tight liner anchored to the concrete.
- Support brackets for the polar crane girder beam.
- On its external side there are three vertical ribs for anchoring the horizontal pre-stressing tendons.
- Bosses and strengtheners around the transfer tube sleeve and equipment hatch.

583 The inner containment cylindrical shell and the dome are pre-stressed concrete structures. Pre-stressing is provided by an arrangement of steel tendons. Each horizontal tendon makes a complete loop of the containment and is anchored within a buttress. Each horizontal tendon is tensioned on both ends. The vertical tendons form two main groups: the ‘gamma’ tendons, and the ‘pure’ vertical tendons. The ‘gamma’ tendons are vertical tendons which are returned to the dome and which are tensioned at

both ends. The upper end is anchored at the dome ring and the lower end is anchored in the vertical tendons pre-stressing gallery, located underneath the support slab. The 'pure' vertical tendons are tensioned at their upper end located in the dome ring and are passively anchored in the gallery beneath the support slab.

584 Each pre-stressing tendon consists of 54 T 15.7 class 1860 cables with a initial tension of 0.8 f_{pk}. The initial force is 12.06 MN/tendon. There are a total of 47 vertical tendons, 119 Horizontal tendons and 104 gamma tendons.

585 The tendons themselves are located in steel ducts (or sheaths). These are either standard ringed sheaths (thickness 0.6 mm) or rigid tubes (thickness 2mm). The former are used for straight or only slightly deviated sections. The bending of the two sheath types is limited respectively to 8 and 10m radius and realised in situ. The connections between sheaths consist of sleeves with a length of 4 times the sheath diameter, both ends being sealed by a thermo-retractable sleeve.

586 In order to avoid any introduction of liquids during concreting or leaks during grouting of sheaths, it has been decided to use rigid tubes for vertical tendons, dome tendons, cables situated less than 5 cm from a concreting joint and for some particular parts of other sheaths. These tubes are bent by a roller machine and widened by a special jack in a workshop on site. For this type, the bending radii are smaller (6 and 8 m). Between two tube elements, the connections consist of a resin adhesion completed by a thermo-retractable sleeve.

587 The tensioning of horizontal cables takes place after the tensioning of the vertical cables in order to avoid excessive flexural effects. The tensioning of the gamma cables is carried out when concreting of the dome has been completed and when the concrete of the last layer has reached more than 28 days and attained a compressive strength value of 60 MPa. This is in order to ensure that the concrete has sufficient mechanical resistance to support the tensioning of the cables and thus limit the concrete creep deformations.

588 Following tensioning, the ducts are injected with a cementicious grout, the intention of which is to fill completely the voids between the tendons and the duct walls. There are time limits placed on the tensioning operation and the grouting operation to ensure a limited exposure of unprotected tendons to potentially deleterious atmospheric conditions.

589 The steel liner plate fully covers the inside surface of the containment structure walls, dome and top surface of the support slab. This continuous membrane provides a containment boundary against which leak-tightness criteria is applied. For this reason, the steel liner plate is located between the top of the foundation raft and the internal structure support slab. The steel liner is designed to ensure leak-tightness under normal operating conditions, during tests on the containment and in accident conditions. The steel liner is used as a form for the construction of the inner containment concrete wall. A continuous anchoring system is integrated into the concrete and welded to the steel liner plate. It comprises continuous steel anchors crossing at right angles to form a mesh. In each of the meshes there are stud anchors. The role of the anchoring system is to stiffen the steel liner plate and ensure stability of the liner during construction and operation. The continuous anchorages transmit concrete deformation to the steel liner plate. They limit the movement of the steel liner plate in case of differences of thickness, temperature or elastoplastic conditions, between two adjacent meshes in the steel liner plate. In addition, they provide the liner with sufficient rigidity during its assembly and during the construction phase. The localised anchorages prevent the grid from buckling. The spacing of the anchorages is such that local bending, which may occur in the steel liner

plate during pre-stressing or when heated, due to geometrical manufacturing defects, remains within acceptable limits.

590 The containment is constructed with a large number of sensors built in, including strain gauges, inclinometers, levelling points, hygrometers, pendulums, temperature probes, invar wires. Additional instrumentation is provided during the decennial pressure tests.

591 The preliminary containment analysis was performed in accordance with ETC-C Revision B. The design of the containment has been undertaken by Coyne et Bellier for EDF and AREVA.

4.3.6.4 Assessment- Design Concept

592 A review of the various Reactor Building specifications and hypothesis notes (Refs EA36 and EA37) is given below.

593 A review of Ref. EA36 (Reactor Building Internals Hypothesis note by Sofinel) has identified the following:

- The document is specific for Flamanville 3.
- The document is very non specific in terms of requirements for a large number of items. This is especially true for some of the design loads.
- The design life of the structure is given as 60 years, which is not adequate, since this does not include an allowance for construction and post operational phases.
- Extensive references are made to the Flamanville 3 PSAR French legislation and decrees as well as standards.
- A number of the key references have been superseded.
- The document needs to reflect the latest position on how load drops will be considered in the design.
- There are no clear requirements to consider robustness or global stability of the NI structures as required by the UK Building regulations Part A.
- There is no reference to the need to consider the CDM regulations or their French equivalent.
- There is a lack of detail in a number of areas including structural philosophy, analysis methods, interfacing with adjacent structures etc.
- It is stated that there is a requirement for the reactor vessel pit to be completely dry, but insufficient guidance is given on how this should be achieved.

594 A review of Ref. EA37 (Reactor Building Internals Hypothesis note by COB) has identified the following:

- The document is specific for Flamanville 3
- The design life of the structure is set at 65 years, with 20 years decommissioning which is adequate.
- Extensive references are made to the Flamanville 3 PSAR French legislation and decrees as well as standards.
- A number of the key references have been superseded.

- The document needs to reflect the latest position on how load drops will be considered in the design.
- There are no apparent requirements to consider robustness or global stability of the NI structures as required by the UK Building regulations Part A.
- There is no reference to the need to consider the CDM regulations or their French equivalent.
- There is a lack of detail in a number of areas including structural philosophy, analysis methods, interfacing with adjacent structures etc.
- For a number of the accident scenarios, the loading is not clearly defined; references are made to future workscopes. This is the case for some reactor pit thermal loads, internal missiles, and pipework rupture.

595 Taking account of all the above comments, it is considered that the suite of documents examined EA36, 37 do not provide sufficient information to enable them to be used in the UK as hypothesis notes for the design of the Reactor Building. This is part of GDA Issue **GI-UKEPR-CE-01**.

596 The containment structure is a pre-stressed, post tensioned structure with the tendons permanently grouted in place following tensioning with a cementicious grout. I had two key observations on this design during the Step 2 review;

- There is no means of conducting post installation checks on the level of pre-stressing remaining in the tendons.
- There is no means of confirming the ongoing integrity of the tendon material through direct inspection of the tendons.

597 These observations were transformed into RO-UKEPR-017 during Step 3, and the responses reviewed during Step 4. Assorted TSC supporting reports Refs TSC5, TSC6 and TSC17 provide a more detailed assessment of aspects of the RO-UKEPR-017 responses.

598 The following sections (Sections 4.3.6.4.1 to 4.3.6.4.4) detail the assessment history of the responses to RO-UKEPR-017. The final position is detailed in Section 4.3.6.4.4, however the earlier assessments are included for completeness. Additionally, they provide a logical explanation of how the final resolution of concerns was reached.

4.3.6.4.1 Regulatory Observation RO-UKEPR-017 First Response

599 Within the UK, all the pre-stressed concrete pressure vessels and containments for nuclear applications have been constructed in a manner which allows routine load testing and removal of tendons for inspection. It is therefore a novel technology in this application in the UK.

600 Report ENGSGC080361 (Ref. EA8) was provided as an initial response to RO-UKEPR-017. The report was reviewed and this highlighted a large number of areas where further information was required and this was agreed with EDF and AREVA. EDF and AREVA Letter EPR00135R and assorted attachments (Refs EA114 to EA120) was subsequently provided. This second response to RO-UKEPR-017 has a much improved coverage of information and greater depth of supporting evidence.

601 A detailed review of Refs EA114 to EA120 revealed a continued lack of depth in the consideration of the response. In order to provide some regulatory clarity to the Issue, I

outlined the basis of a claims argument evidence arrangement which I would expect to see in support of the RO-UKEPR-017 response.

602 The two key claims are as follows.

- The design provides adequate reliability through the life of the structure.
- There are no reasonably practicable modifications that can be made to improve the design.

603 The key arguments which support these claims are as follows.

- Design methods are robust.
- Design is capable of being implemented.
- Design offers adequate protection against corrosion.
- Design has redundancy.
- Degradation (i.e. non-predicted) is detectable.
- Beyond Design Basis behaviour is predicable and ductile.

604 In addition, a suitably detailed ALARP review is required.

4.3.6.4.2 Regulatory Observation RO-UKEPR-017 Second Response

605 EDF and AREVA provided some revised responses to RO-UKEPR-017 in Letter EPR000272 and attachments, Refs EA121 to EA127. The response was not considered satisfactory, and following meetings in March 2010, the RO-UKEPR-017 was extended to include 10 specific actions. These covered the following areas

606 RO-UKEPR-017.A1 concerned information on the proposed grout formulation for use in the UK EPR (Ref. EA124). A more complete overview of the grout specification is required.

607 RO-UKEPR-017.A2 concerned information on previous mock ups used on the Civaux site and some indication of likely mock ups for the Flamanaville 3 and Olkiluoto sites. A more complete overview of the likely mock ups to be undertaken in the UK as well as the acceptance criteria is required. (Ref. EA124)

608 RO-UKEPR-017.A3 concerned the assumptions made in the Coyne et Bellier Study (Ref. EA125) on the re-anchorage lengths of tendons that there will be no change to the re-anchorage length should a demand be placed on the containment. Further justification is required

609 RO-UKEPR-017.A4 requested a rationale for the selection of tendons used in Coyne et Bellier report (Ref. EA125) which examines the sensitivity of the containment to loss of pre-stressing and examines the capability of the strain monitoring system to detect any losses.

610 RO-UKEPR-017.A5 concerned the reconciliation of the conservative assumptions made over the .tendon re-anchorage lengths and the resulting non-conservative results for tendon loss detection to provide a coherent safety argument.

611 RO-UKEPR-017.A6 concerned Ref. EA123 which contained some information on the monitoring of the containment. It does not contain sufficient information on the link to the safety justification, the periodicity of measurements, the alert/ trigger levels and subsequent action plans.

612 RO-UKEPR-017.A7 concerned Ref. EA123 which contained some information on the 10 yearly pressure test of the containment. It does not contain sufficient information on the link to the safety justification, the acceptance criteria , the alert/ trigger levels and subsequent action plans.

613 RO-UKEPR-017.A8 concerned a lack of clarity over the rationale and choice of instrumentation selected for inclusion in the DAO.

614 RO-UKEPR-017.A9 requested that the ALARP justification provided in Ref. EA122 be further developed. There is no consideration of potential modification to the design/ construction/ monitoring/ testing regime for the containment.

615 RO-UKEPR-017.A10 requested further justification for the time periods between threading of the tendons and tensioning and between tensioning and grouting.

4.3.6.4.3 Regulatory Observation RO-UKEPR-017 Third Response

616 The response to these ten RO Actions was received between August and September 2010 (Refs EA131 to EA137). Detailed review work was undertaken on these documents (Refs TSC16 and TSC17). The key outcomes from this review are repeated below.

RO-UKEPR-017.A1 – Grout Formulation (Appendix to Ref. EA128)

617 There are some residual issues which needed some further clarity :

- Analysis of Bleed water.
- Delays mid way through grouting.
- Measurement of grout volumes.
- Use of grout flowmeters.

The choice over whether to use of pre-bagged or site mixed grouts is a topic which can be left to a site specific stage.

RO-UKEPR-017.A2 – Use of Mock ups (Appendix to Ref. EA128)

618 There was still an unacceptable level of vagueness in the acceptance criteria and on the sequence of acceptable tests that are sufficient to demonstrate that grouting can be achieved. Typical examples are:

- What is meant by partial mock ups?
- The use of previous data from other tests is an area I would be uncomfortable with.
- Void sizes are ill defined.
- *“potentially damaging”* cracks - are not defined.

RO-UKEPR-017.A3 – Anchorage lengths of tendons (Refs EA131, EA132 and 133)

619 The regulatory observation was concerned with the justification for the anchorage length used in the assessment of the strength of the containment with broken tendons. Originally, the anchorage length was based on the pre-stress force not considering any increase due to pressure loading. The response showed that the increase in force due to pressure loading was relatively small. Realistically, it is well within the accuracy of prediction of anchorage lengths.

620 The assumptions over very long anchorage lengths (up to some 20m) imply movements of the order of 50mm of the free end of the tendon to ensure strain take up. It is not clear the bond behaviour is ductile enough to work with such large movements, i.e. it could potentially unzip. Although this question is quoted in the report, there does not appear to be an answer.

RO-UKEPR-017.A4 – Tendon Selection Criteria) (Refs EA134 and 135)

621 The response partially addresses one of the questions on "broader applicability" the issue of criticality of the tendons examined is given scant coverage and the issue of greater likelihood of degradation not discussed.

622 The response is still essentially justifying the choice of tendons as being representative. Although this answers expectation 1 ("it is not clear what the rationale for the selection of tendons was") it merely says this was the reason, rather than justifying it. It does not address the expectation that selection would be based on criticality, broader applicability, or increased likelihood. The result is that the report still shows how many adjacent typical tendons can be lost without reducing strength below the required design strength, without answering the rather different question of what is the minimum number of tendons that can be lost without this problem. It still appears possible, for example, that a smaller number failing adjacent to a large penetration (such as the equipment hatch) would reduce strength by the same amount due to lack of tendons or reinforcement on one side of the breaks to redistribute the force to. This is particularly concerning as the tendon deviation makes this a particularly likely place for voids in the grout.

RO-UKEPR-017.A5 – Re-anchorage lengths (Refs EA136 and 137)

623 Previously, the assumed area over which a tendon break could be detected by strain gauges was estimated from an anchorage length derived from EN 1992 which could be as long as 20m. According to EDF and AREVA this has now been "drastically" reduced to 4m which is a significant improvement: 4m does indeed sound a "drastic" reduction on the EN 1992 calculated length. However, anchorage lengths are known to be very variable and EN 1992 is looking at anchorage lengths for cases where long is conservative, whereas for this case short is conservative. Furthermore, the Poisson's ratio effect reduces bond strength when a force is applied to the tendon but increases it when a force is released. There is some evidence of much shorter lengths of re-anchorage (Length over diameter ratio scaling from smaller tendons) there is no evidence directly applicable to tendons of this size. It is considered that whilst there could be situations where the re-anchorage length is lower than 4m, that further reductions of the anchorage length for studies on detection thresholds are not going to yield significantly more useful information.

RO-UKEPR-017.A6 – Monitoring (Appendix to Ref. EA128)

624 The response is basically a repetition of current EDF and AREVA (DTG) practice. There is no link to the safety justification, despite this being the main thrust of the RO Action. Additionally, the approach examined the case where wholesale failure of a tendons occurs, which clearly introduces a step change in the strain state of the containment. In reality, a more gradual degradation of the tendons over a protracted period is more likely. The emphasis would therefore need to be on a greater review of trending of results. This would require a more frequent recording of data than currently proposed. In addition, the practicability of isolating tendon degradation effects on the strain state from creep and shrinkage would need to be developed.

625 The process of derivation of trigger levels and the subsequent alerting arrangements would need to be developed more fully for the UK.

RO-UKEPR-017.A7 – Pressure Testing (Appendix to Ref. EA128)

626 The report describes the monitoring of the pressure tests. It says it is checked that strains are linear, reversible and within 30% of expected. Once the effective elastic modulus has been deduced from the monitoring during pre-stressing, a 15% criterion is used. In practice, since the vessel has to be stressed before pressure testing, this appears to mean the 30% criterion is applied to the stressing and 15% to the pressure test and it is not clear why it is not written that way. It might be to allow for the possibility of the monitoring from the stressing not having been analysed at the time of the pressure test which appears undesirable. The monitoring of the stressing should be required to be analysed before the time of the first test, in which case the criterion should then be tightened to 15%.

627 Reversibility is expected to be within 80% although this criterion is not applied to the raft due to irreversible behaviour of the soils and also to the very low strain anticipated. Presumably, the within 15/30% of predictions criterion is also not applied to the raft since they do not appear to have any predictions for the raft.

628 Within an essentially axi-symmetric structure, lack of symmetry could potentially give an earlier warning of local problems such as with loss of tendons and this should also be considered. By the present criteria, a difference between strain gauges symmetrically located and predicted to give identical readings of up 35% (85% using the 30% criterion) could arise without raising any alarms.

RO-UKEPR-017.A8 – Instrumentation (Appendix to Ref. EA128)

629 The approach has essentially shifted the detailed evolution of the instrumentation into site licensing. There is little that can be disagreed with, however it should be recognised that considerable work will be required. There is little description of how this will be achieved and how it will link to the safety case and analysis.

RO-UKEPR-017.-A9 – Overall safety justification (Appendix to Ref. EA128)

630 It is not that clear what the key claims that are being made are. They are scattered across the document. In addition, there are inconsistencies in this document with others in the RO-UKEPR-017 suite.

631 The comparison between bonded and unbonded is clearly biased and not consistent with some of the other findings. For example it is claimed that bond “helps prevent propagation of rupture”. Given the anchorage lengths found in the other studies any benefit is marginal and indeed unproven.

632 The arguments for grouted are largely repeated from previous reports and will not be fully reconsidered here except to note that they mix economic and safety issues so that some are not strictly relevant to the ALARP principle. There is no real link to the beyond design basis work presented in the response to RO-UKEPR-037.

633 The report also questions whether re-tensioning in structures with the complexity of the EPR is effective, suggesting “for horizontal and gamma tendon re-tensioning might only affect a small length of tendon close to the anchorage.” No details of the source of this concern are given. It does not appear to be either consistent with UK experience or theoretically explicable. With the low friction coefficients normally observed with unbonded systems, re-stressing of even very highly deviated tendons should be effective.

Indeed, it is not clear why it should be any less effective than initial tensioning. One possible explanation is that the bad experience is based on attempts to re-stress using a strand (rather than cable) jack. This can lead to problems with friction due to the strand being stressed being pressed against the inside of bends by other strands, a problem which is avoided if all the strands are stressed together.

634 The statement that selection of a small sample of tendons in unbonded systems for testing gives limited confidence is misleading. The arguments over lift off tests and the level of effort required are only partially valid.

635 The statements in the ALARP section on containment monitoring, use of acoustic monitoring and use of grouting flowmeters are not well developed and require further justification. The ALARP section on timings between construction options needs to be viewed in conjunction with Action 10. It is not clear why such large "batches" are necessary.

636 The comments above on the RO Action responses were passed over to EDF and AREVA as part of a two day workshop on 8th and 9th of December 2010. Following that meeting, EDF and AREVA undertook a series of actions to upgrade some reports and to provide additional supporting information. This was done under cover of letter Ref. EA138, which provided Refs EA139 to EA147.

4.3.6.4.4 Regulatory Observation RO-UKEPR-17 Final Response

637 This provides a review of the final responses from EDF and AREVA on RO-UKEPR-017. This provides the final agreed position prior to closure of RO-UKEPR-017. The concerns identified in Sections 4.6.4.1 to 4.6.4.3 have all been addressed with the exception of those noted in this section which are to be dealt with in the licensing phase and are identified as assessment findings.

638 A review of the responses to the revised response are provided below. They are not focussed on the individual RO Actions, but derive from the primary document provided for RO-UKEPR-017.A9 (Ref. EA139) which acts as a signpost to a number of other documents.

639 Ref. EA139 is a revised version of Refs EA127 and EA130. It has been provided with added clarity, and the claims argument evidence structure is much clearer. The paragraphs below discuss the response in more detail and identify any remnant concerns which exist.

640 The main aspects of the safety justification presented in Ref. EA139 are repeated below.

- Design and construction methods are reviewed to confirm that a robust design can be achieved which can be constructed to high standards of quality and reliability.
- The tendons are shown to be adequately protected from corrosion during installation and subsequent operation of the plant. Feedback experience from existing NPP containments constructed using grouted tendons is presented to confirm that no evidence of corrosion has been found.
- Results of Finite Element modelling are presented which show that the containment structure is tolerant to multiple failures of entire tendons, however unlikely, even for the extreme case where the failures occur in close proximity inside the containment wall.

- Monitoring of pre-stressing by periodic pressure testing and in-service measurements of concrete strains is shown to provide confidence that any degradation of pre-stressing would be detectable before it threatened the resistance of the containment.
- The design is shown to give high margins of safety for beyond design accident conditions by analysis and from results of test programmes in a containment mock up.

641 The main aspects presented above are examined in more detail in the following.

642 The design standard used for the containment is the ETC-C. The detailed design documents examined have been based on the Revision B of ETC-C. This version has not been found to be entirely satisfactory for use in the UK. However, it is considered that the 2010 AFCEN version with the UK companion document when complete will provide a suitable framework for the design of the containment.

643 The computer programmes used to support the design have been examined and found to be suitable (see Section 4.2.4).

644 The issue of reliability is however still an open one. RO-UKEPR-037 was raised on this topic and is discussed separately in this report.

645 The specification of grout, use of mock-ups, timing of operations and the monitoring of the process for grouting have all been claimed as adequate in Ref. EA139. The grout specifications now refer directly to the relevant part of ETC-C which quote Euronorms EN 445 to 447 (Ref. OD43-45).

646 A recommendation to analyse the grout bleed water for corrosive constituents has been argued as unnecessary because the bleed is quickly reabsorbed into the grout. Unfortunately it is now recognized that bleed water can be present for long enough to cause degradation. Moreover there have been cases where fractured strands have been identified as having been caused by bleed water. A chemical analysis by the supplier of pre-bagged grout provides assurance and is relatively quick and easy to carry out at the time when the grout is formulated.

647 Within the UK, it is standard practice to use pre-bagged grout rather than site batched. The current proposal for the UK EPR still says pre-bagged grout will be used if suitable material is available. Whilst it has now been accepted that approval of the final choice between site mixed and pre-bagged grout will be deferred to the site licensing phase, the wording appears to exclude the possibility of developing a new pre bagged grout which would be practical for a project of this scale. There will be a need to ensure that any proposed site batched grout achieves a level of consistency and quality equal to or better than a prebagged grout. The current grout specification will need to be developed as part of the site licensing phase to include all the comments listed above (Assessment Finding **AF-UKEPR-CE-32**).

648 There is an implication that grout pumping could stop for up to an hour as it says new tests are required if this arises. It is surprising that such a delay is considered acceptable and it is considered that it would normally be very difficult if not impossible to re-start after such a delay. It is not clear whether it means stopped between ducts or during grouting of a duct. With thixotropic grout it would be impossible to re-start grouting of a duct after such a long delay. However, it is considered that even a delay of an hour between ducts whilst using the same batch of mixed grout should be avoided. Even without such apparently unnecessary delays it appears to be necessary to use retarders to avoid premature hardening of the grout. The detailed review of the proposed grouting methods will be undertaken at a site specific level and will need to reflect on the chosen grout specification.

649 Current methods of measuring grout volume are crude and flowmeters offer a great improvement. Flowmeters are easy to use, accurate, and provide numerical records. They have been used successfully on site albeit contractors with no experience of them may be fearful of the unknown. The purpose of their use is not to detect very small voids. It is to confirm that grout injection has not run into problems such as leakage into other ducts at crossovers as has occurred in nuclear structures or generation of gross voids. I therefore welcome the use of flowmeters whilst accepting the arguments given that they do not give sufficiently accurate volume information to identify small voids.

650 I understand the partial mock-ups are at full *scale*. It actually says here *"the term full scale mock-up applies only to pure vertical and highly deviated horizontal cables (scale 1:1 and full length).* This might be read as implying the gamma cable mock ups are not full scale but in fact we understood they are in two parts but each part is full scale.

651 The acceptance criteria for the grouting mock-ups are not considered sufficiently detailed at present. For grouting caps, it is implied that voids are acceptable irrespective of size provided caps, wedges, anchor heads and strands are not visible irrespective of the thickness of the covering grout. For vertical ducts, it is implied that no void, however small, is acceptable. For all cables, it is implied that no detectable longitudinal crack, irrespective of how narrow, is acceptable. For each of these criteria, a more practical approach is needed which will give a clearer definition of what is acceptable.

652 The use of previous data for qualifying methods and approach for the first UK EPR is not something I would accept. It is still not clear what procedures will be used if one of the mock-ups fails to meet the required criteria. Taken literally it only says 3 successful mock ups are required, implying they could be, say, numbers 3, 5 and 7 of 7. This would clearly be unsatisfactory and some clarification will be required at a site specific level.

653 In summary, there is a need to provide as part of the site licensing phase a more robust set of specifications for the mock ups and the acceptance criteria. These shall be clearly and unambiguously written to ensure that it is clear when successful trials have been completed (Assessment Finding **AF-UKEPR-CE-33**).

654 The timing of grouting operations is key to ensure that minimum exposure times of strand to potentially corrosive environmental conditions, typically chloride laden atmospheres for coastal locations. The existing criteria laid down by EDF and AREVA are that tensioning must place within 4 weeks of threading and that grouting within 2 weeks of tensioning. Times achieved from Civaux during construction are c 12 days and 9 days respectively. During the construction of Olkiluoto 3 similar times have been achieved.

655 EDF and AREVA claim that the timing of the operations is a result of the use of pre-stressing phases which include between 11 and 43 tendons. Within each phase, grouting cannot begin until the verification of the entire phase of tendons is complete. Whilst these arguments are agreed in principle, we remain to be convinced that the limitations placed in the specifications cannot be reduced, whilst still maintaining a practicable approach. This is borne out by the experience from Civaux and Olkiluoto. It is therefore an assessment finding that the tensioning and grouting timing restrictions should be validated on a site by site basis. The licensee shall develop tensioning and grouting timing restrictions on a site specific basis (Assessment Finding **AF-UKEPR-CE-34**).

656 The initial submission of the design of the containment provided little information on the margins available in the design. Analysis on the main containment vessel wall and on the area around the equipment hatch have been undertaken (Refs EA125 and EA141). These analyses essentially removed the effects from tendons and compared the resulting stress fields against the requirements of ETC-C. The ETC-C requirements essentially

limit the behaviour to the linear range, and the figures for the number of tendons which are required to fail can be considered to be lower bounds. For the most critical load cases, 6 horizontal tendons and 3 gamma tendons are required to fail. However this is against group 2 criteria, which have even smaller allowables. The analysis against the 0.65 MPa case (group 3 allowables) [see Section 4.3.6.5 for definitions] has values of 7 horizontal tendons and 4 gamma tendons. Additionally, it should be made clear that these tendons are assumed to be immediately adjacent to each other rather than uniformly distributed. It is therefore clear that there is considerable margin against breach of the design case for the rupture of individual tendons.

657 The initial calculations of the tolerability of the design were based on the calculations in EA125, which considered only horizontal tendons in typical zones of the wall and dome. Arguments were developed by EDF and AREVA to try and argue the greater applicability of the findings to special zones, such as around openings through the containment. I discounted these arguments as lacking sufficient technical rigour and EDF and AREVA undertook work on the loss of tendons around the Equipment Hatch. This work (REF. EA141) has shown that there is not a significant difference in the criticality of this zone to tendon loss compared to typical zones in the wall; or the dome. I am satisfied that I now have a good understanding of the sensitivity of the vessel to tendon losses, and that the structure has a sufficient level of tolerability.

658 The corollary to having a lack of sensitivity to tendon loss is that the detectability of tendon losses needs to be well understood. The same models used for the examination of tendon losses on the structure as a whole were used by EDF and AREVA to predict the changes in strain fields surrounding tendon losses. These predicted strain changes are then mapped over the strain gauge layout to ensure that the level of strain change that can be detected occurs well before the number of tendons lost are sufficient to challenge the design basis.

659 The calculations for the strain effects have been undertaken for two basic scenarios relating to re-anchorage length. Prediction of the actual re-anchorage length is not straightforward, and code approaches are not that helpful. Ref. EA141 uses what it calls *“an artificially small re-anchorage length (4m compared with a value of 12.7 m calculated by Eurocode methods)”.* As noted earlier, the comparison with EN 1992 is somewhat misleading. That gives a conservative anchorage length assuming that long is conservative whereas for this purpose short is conservative. Because of the variability of anchorage lengths and the fact that the EPR situation (with high cover, heavy reinforcement and orthogonal pre-stress) could give good anchorage conditions, it seems likely anchorage lengths could be shorter. Some tests on structures where tendons have been cut give lengths of 1 to 2m. In all respects except size of tendon, the EPR conditions are closer to the former. Correcting for scale (i.e. assuming anchorage length is fixed relative to tendon size) suggests an anchorage length of 2m may be possible for the EPR. Thus whilst 4m maybe a reasonable length to assume, it is not at all clear claiming it is “*artificially small”* is justified.

660 A related question is around the mechanics following tendon rupture. When tendon anchorages are designed, they have to cope with very high “bursting forces”. If a tendon were to breach, as part of its re-anchorage bursting forces would develop around the tendon. TQ-EPR-1334 requested further information on this topic. The effect of bursting force on capacity is considered for two cases, under the 0.55MPa (group 2 accident). The first capacity considered is broadly consistent with the group 2 properties except the pre-stressing is assumed to be 100% effective rather than 40% effective. It is shown with these assumptions that after 2 tendons break the strength criteria is just acceptable. The second capacity is effectively a group 3 check with the liner yielding in tension, with this

criteria they show that the capacity is just acceptable with 3 tendons broken. To put in context they need to adopt group three strength criteria to satisfy a group 2 load with three tendon breaks. This check is only carried out in the vertical direction.

661 It is probable that the method used in the original calculations and applied above is conservative, and that assuming a lower strut angle would reduce the bursting forces. In addition any splitting would reduce bond and increase bond length reducing bursting forces. However in my opinion delamination around a tendon break is a valid concern. It is less clear what the consequence of this delamination would be but effects would include reduced shear capacity for other loads, and reduced bond and therefore contribution to limiting stresses and crack widths for any other tendons in the delaminated zone. On the positive side it would make detection of tendon break more likely.

662 The understanding of bursting forces around broken tendons is not fully developed. There is therefore a need for the licensee to investigate the effects of bursting forces on the integrity of the containment wall and the effects of subsequent strain predictions (if any) incorporated into the proposed layout of containment instrumentation (Assessment Finding **AF-UKEPR-CE-35**).

663 The assumption is that whole tendons fail at a time and that a strain change of 10^{-5} would be not only detectable but detected and distinguished from other strain changes. Putting these issues together it seems clear that whilst the instrumentation does give a *probability* of detecting the loss of pre-stress quoted, it is unrealistic to attach absolute certainty to it. What has not been fully developed as yet are detailed arrangements for the frequency of monitoring and alerting and subsequent actions to be taken. This is an assessment finding that detailed arrangements will need to be developed by the licensee for the management of the monitoring of the inner containment (Assessment Finding **AF-UKEPR-CE-36**).

664 The strain gauges proposed are vibrating wire type, which have been in common use in large concrete structures for many years. In the UK nuclear sector, we have wide experience of these gauges and are comfortable with their reliability and longevity. The layout of gauges originally proposed has changed markedly as a result of the discussions over the effects on vessel strain as a result of tendon loss. This has forced a more regular grid of gauges. The layout of the gauges in principle appears to be satisfactory. In Ref. EA144, there are some rather broad statements that there may be a need to have extra gauges for other reasons. It is assumed that this is in relation to confirming the predictions of mechanical behaviour in key areas. I would wish to examine the layout of the proposed approach on a site specific basis however as the details provided thus far are relatively broad in nature. This is an assessment finding. The licensee shall develop the layout of the containment instruments sufficient to satisfy the requirements of the PCSR (Assessment Finding **AF-UKEPR-CE-37**).

665 Ref. EA143 describes the monitoring of the pressure tests. The report is very similar to Ref. EA128 and the same concerns are expressed over the criteria expressed within it (See paras 627 - 629).

666 Visual inspections are also undertaken and cracks should not be wider than 0.6 mm. It is not entirely clear what happens if wider cracks are observed but presumably they would be investigated.

667 Overall the approach appears reasonable. However, 15% gives quite a wide margin for error and one might have expected the behaviour of a structure which is within the elastic region and which is made of a known concrete to be closer to predictions. The use of automatic monitoring systems, may make it necessary to explicitly include alerts for such

effects: such a difference would most probably have been noticed using a manual system even without any specific instruction to look for it. An alternative is to require the readings (or, perhaps even more revealing, the difference between readings and predictions) to be plotted on a contour plot and reviewed by an Engineer. This has the advantage of showing up other oddities which might not have been identified in advance as specific alerts.

668 The arrangements described in Ref. EA143 are also heavily dependent on the DTG specifications and practices. Whilst these are not necessarily incorrect, there will need to be a defined set developed for use in the UK.

669 In summary, the criteria for accepting the decennial pressure tests are not sufficiently well developed at this stage in terms of criteria, alerts and acceptance. These will need to be developed as part of the site licensing (Assessment Finding **AF-UKEPR-CE-38**). Furthermore, The licensee shall undertake analysis of the containment structure to reflect the actual concrete properties used in the construction. This shall be undertaken ahead of the containment pressure test to provide a credible comparison for the measured values (Assessment Finding **AF-UKEPR-CE-68**).

670 The ALARP section considers possible improvements to the pre-stressing system and its monitoring. It accepts the increased instrumentation required by the earlier analysis of detection of tendon breaks. It considers acoustic emission monitoring (AE) but dismisses this on the grounds of “complexity and uncertainty of effectiveness” adding that “meaningful interpretation of such a complex system would require expert analysis”. In particular saying the 1992 ASME code does not suggest using it is not relevant as the technology scarcely existed in 1992. It also says that each of the 260+ cables would have to be equipped with acoustical sensors to give equivalent coverage to the strain gauging which it suggests covers the whole pre-stressing system However, this appears to be looking at it as an alternative rather than an addition. It is also not clear its discussion of AE is consistent with current practice which does not normally use separate monitors for each cable and which uses filtering systems which have proved able to identify sounds due to wire breaks remarkably well including in noisy environments. The references it quotes are all either old or are supportive of the use of AE as a technology. AE is a new and still developing technology. .

671 Overall EDF and AREVA appear to be comparing a pessimistic view of AE with a view of the proposed strain gauging which is relatively optimistic. Whilst it is probably correct that the approach would not give certainty of detection, it is much less clear that it would give lower certainty than the proposed strain gauging system. Further, within the ALARP principle it is not at all clear the comparison of the two approaches is even relevant. It is clearly not certain the strain gauges would detect small numbers of whole tendon failures let alone wire failures. The AE approach may do this. The two approaches are complimentary rather than in competition. The argument that the approach should not be relied on may be right but using it as an additional detection system does not constitute relying on it. At the time of site licensing the licensee will need to review available options for using Acoustic emissions and other non invasive detection systems on the inner containment structure as could be available to them (Assessment Finding **AF-UKEPR-CE-39**).

672 The possibility of increasing the frequency of pressure testing is reviewed in Section 4.1 of EA139. A quantitative analysis of the potential effect of this on reliability is presented. This relies on an assumption of probable failure rate of tendons from Appendix 1 which is disputed. However, the conclusion is seen to be valid but it is not clear the justification

given is. The use of decennial full scale pressure tests is in line with international best practice.

673 Possible changes to improve resistance to corrosion are considered in Section 4.2 of Ref. EA139. Individually sheathed strand are dismissed. Greased systems have the same disadvantage of other unbonded tendons and issues of lack of experience and lack of heat resistance of the strand are used to dismiss the system. Similar arguments are used to dismiss HDPE ducts. It is not clear if any effort has been made to see if it is possible to develop more heat resistant materials.

674 Possible improvements in construction are considered in Section 4.3 of Ref. EA139. The possibility of reducing the times from threading to stressing and stressing to grouting is considered. It is concluded that this is not possible because of the need to stress in groups and review findings before grouting. The principle of this is clear but the reason for the size of the groups is not. These vary from 14 to 43 tendons. It seems implausible that it is not possible to split the group of 43 into smaller subsets. Assessment Finding **AF-UKEPR-CE-34** has already been raised to address this on a site specific basis.

675 As part of the submissions for RO-UKEPR-037, a study on the reliability of the containment was undertaken. This included an examination of the beyond design basis performance of the containment. The initial submission for this work was not considered adequate, and a revised response has been provided, however there has been insufficient time to complete an assessment. This has been raised as GDA Issue **GI-UKEPR-CE-05**.

4.3.6.4.5 Assessment- Detailed Design

676 The detailed design of the containment has been undertaken by Coyne et Bellier (pre-stressed concrete wall) and NFM (liner and penetrations). The design is an evolution of previous containments in France. Ref. EA30 provides an overview of the design evolution as well as a route map through the design process, identifying the key documents.

677 Refs EA31 to EA35 are the key hypothesis notes for the containment.

678 A review of Ref. EA31 has identified the following:

- The document is Flamanville 3 specific.
- There are references to baseline safety requirements and preliminary safety requirement documents.
- There are references to French public procurement rules and other French standards.
- The guidance on the construction of the finite element models for the structure is insufficient and would need to be supplemented by other guidance.
- There is detail on load combinations and replication of aspects of the ETC-C which may not fully align with the 2010 version of ETC-C and the UK companion documents requirements.

679 A review of Ref. EA32 has identified the following.

- The documents are Flamanville 3 specific.
- There are references to French public procurement rules and other French standards.

- There is detail on load combinations and replication of aspects of the ETC-C which may not fully align with the 2010 version of ETC-C and the UK companion documents requirements.

680 In summary, documents Refs EA31 and EA32 requires a substantial would require a substantial revision to allow their use in the UK as a detailed structure specific hypothesis documents. This is part of GDA Issue **GI-UKEPR-CE-01.**

681 As part of the overall safety approach implemented at the design stage of the EPR project, the civil engineering structures must fulfil a dual function. On the one hand, they must protect the plant against all the hazards to which it might be exposed, in particular external hazards. On the other hand, they must protect the environment against all the accident conditions that cannot practically be eliminated, and in particular restrict the release of radioactive material in the most severe conditions. The design of the containment is thus based on consideration of various loads and 'design basis conditions' determined by taking into account the following events:

- design basis operating conditions (PCC1 to PCC4);
- multiple-failure operating conditions (RRC-A) and accidents with core melting (RRC-B);
- internal hazards (HELB, internal flooding, internal projectiles, falling loads, fire, etc);
- external hazards (earthquakes, aircraft crash, external explosion, external flooding, increase in the level of the water table, exceptional meteorological conditions, etc); and
- conditions analyzed for defence in depth that might lead to margins being incorporated into the design of civil engineering structures.

682 These functions are defined as follows:

- A_B serviceability of concrete walls. Application of stresses resulting from a given load must not alter the subsequent behaviour of the structure throughout its lifetime. The structure must remain serviceable with regard to the use for which it was designed.
- A_m: serviceability of metal structures. Application of stresses resulting from a given load must not alter the subsequent behaviour of the structure throughout its lifetime. The structure must remain serviceable with regard to the use for which it was designed.
- R_B: capacity of the concrete wall to withstand the applied load. Permanent deformations can be accepted if the structure concerned remains stable and retains its capacity to support the equipment associated with it.
- R_m: capacity of the metal structure to withstand the applied load. Permanent deformations can be accepted if the structure concerned remains stable and retains its capacity to support the equipment associated with it.
- E: Fluid retention leaktightness. Leaktightness must be ensured under all conditions, even if the reservoir concerned might undergo permanent deformations.
- C: capacity to contain radioactive substances. This mainly concerns the inner containment wall, for which a leakage rate criterion must be complied with.

683 These functions are linked to the conditions as outlined in Table 5 below.

Table 6: Summary of Design Conditions and Criteria for the Inner Containment

Conditions: Categories and Definitions		Inner Containment Wall	Steel Liner	Penetrations
N	Reactor state A to F ambient conditions	A_B	$C+A_m$	$C+A_m$
E1	PCC2 ambient conditions	A_B	$C+A_m$	$C+A_m$
E2	Inspection earthquake	A_B	$C+A_m$	$C+A_m$
E3	Snow and wind	/	/	/
E4	Exceptional temperatures:	/	/	/
E5	Water table	/	/	/
E6	Periodic tests and trials	A_B	$C+A_m$	$C+A_m$
A1	Earthquake	R_B	C	C
A2	Aircraft crash	/	/	/
A3	Explosion/Fire	/	/	/
A4	HELB/Projectiles	R_B	C	R_m
A5	PCC3/4 and RRC-A ambient conditions	R_B	C	C
A6	RRC-B ambient conditions	R_B	C	C
A7	Break 2A - LOCA	R_B	C	C
A8	LOCA + Earthquake combination	R_B	C	C

684 The various load cases to be applied to the inner containment wall are defined by the ETC-C:

- construction condition (1)
- normal operation (2)
- inspection earthquake (4)
- water level (5)
- tests (6)
- LOCA (7)
- severe accident and conservative severe accident (8a and 8b)
- HELB or LOCA (8)
- DBE or vibration due to aircraft crash (11 and 12)
- LOCA + DE (16

685 The conditions are assigned into groups which have different allowable values for the concrete and liner elements of the structure, as listed below. This is provided in much more detail in the ETC-C.

- **Group 1 conditions:**
  - construction condition (1)
  - normal operation (2)
  - tests (6)
  - inspection earthquake (4)
- **Group 2 conditions:**
  - internal accident (7, 9)
  - severe accident (8a)
  - DBE and vibration (11 and 12)
- **Group 3 conditions:**
  - conservative severe accident (8b)
  - LOCA + DBE (16).

686 Each group has a set of limitations on strain and stress on re-bar and concrete. The group 1 limitations are serviceability limit states, the group 2 and 3 are essentially ultimate limit states.

687 I have reviewed the approach used, and notwithstanding the ongoing discussions on load factors within ETC-C am satisfied that the approach is reasonable.

4.3.6.4.5.1 Inner Containment Detailed Analysis of Wall Section

688 During the assessment phase, I have focussed on the interface between the liner / penetration design and the design of the load carry part of the containment. It is this interface which is the most complex,

689 Refs TSC18, TSC19, TSC20 and TSC30 provide more detailed evaluation of aspects of the containment design.

690 The inner containment wall is a domed cylindrical structure in pre-stressed concrete fitted on the inside with a steel liner which encloses the reactor internals. With the outer reinforced concrete containment wall it provides the Reactor Building Containment. These two containment walls are separated by a space forming a continuous cavity where the recovery and filtering of potential leaks is possible.

691 The inner containment wall is fitted with a number of electrical and mechanical penetrations, the largest and most important of which is the equipment hatch, which allows the largest reactor coolant system components to enter the containment.

692 The inner containment wall is designed to withstand any increase in pressure inside the reactor building due to an accident situation, with the steel skin providing a leak-tight seal. The principal geometric properties of the reactor building inner containment wall are:

693 Preliminary design of the tendon requirements was undertaken by hand calculation. The topographic distribution of those tendons was then done via the PRECONT software resulting in 47 purely vertical tendons, 119 horizontal tendons and 104 gamma tendons. The results of this phase are detailed in Ref. EA164.

694 Ref. EA30 provides a complete overview of the process, which is not repeated here. The loads from the various models are then combined in line with the ETC-C specific

combinations, and the necessary checks against the group 1, 2, 3 criteria undertaken. This is done automatically for the typical zones by the code FERRAIL.

Table 7: Primary Dimensions of Inner Containment

Measurement Description	[m]
Inside radius of the cylinder	23.4
Thickness of the cylinder wall	1.3
Thickness of the dome roof	1.0
Inside radius of the spherical part of the dome roof	32.0
Inside radius of the torispherical part of the dome ring zone	8.0
Internal volume (approximate)	80,000 m3

695 Singular zones are defined by EDF and AREVA as areas where geometry and loads introduce localised effects in the wall, include the gusset, dome girder, equipment hatch, personnel hatches and polar crane brackets.

696 The basic requirement of the concrete inner containment wall is to contain internal pressure arising from an accident, and to withstand all other loads. This is achieved through pre-stressing.

697 Pre-stressing of the inner containment is designed to ensure there is sufficient wall strength to resist an absolute internal pressure of 0.65MPa under test and accident conditions. The principal design criteria is that under this pressure, but with no thermal loads, the concrete remains in compression across the full width of the containment throughout the life of the plant. To maintain compression design calculations considered the level of stress introduced into passive reinforcement and the steel liner and allowed for redistribution of stress through time due to concrete creep and shrinkage. Passive reinforcement is designed to resist moments introduced by temperature gradients caused by thermal loads during accident conditions adhering to tension and cracking criteria

698 The preliminary design undertaken by COB is documented in Ref. EA116 for the clear areas of the wall and dome and Ref. EA117 for the major singular zones. A global FE model was first developed of the inner containment during preliminary design using FE software COBEF v4.2. It was used to check stresses and determine reinforcement requirements for Groups 1, 2 and 3, and fundamental ULS load combinations. Ref. EA117 Sections 4 and 5 describe the work undertaken for the dome base design and the equipment hatch. A second FE model was developed, an axisymmmetric solid representation of the gusset zone, and used to determine the dimensions of the tapered gusset. This work was carried out using FE software ANSYS v9. Three variants were developed of the ANSYS model (see table below); with steel liner, with steel liner below the gusset base and equivalent forces to simulate fully plasticized liner behaviour; a thermal model for steady state calculations. In all cases the base of the containment was assumed to be rigid. It appears both of these early FE models have formed the basis of the modelling methodology moving forward to the detail design. Consequently, the preliminary design is not considered any further in this report.

699 A 3D solid FE model has been developed of the whole inner containment including principal singular zones using FE software ANSYS v10. Used for analysis during detailed design this model is linear elastic and used for static or equivalent static loads.

700 Two variants have been developed to analyse the mechanical behaviour of the containment wall, as described in Ref. EA116 Section 3.1, each with two sets of materials for long-term and short-term behaviour. A further variant is introduced for accident thermal action. Table 7 below provides further information.

Table 8: Summary of Containment Models

<table>
<tr><td rowspan="3">Model 1</td><td>Long-term</td><td rowspan="2">Represents all the concrete walls and only includes the steel liner as an elastic membrane on the cylindrical gusset (face above - 4.35 m and riser between -7.85 and -4.35 m). Over all other internal concrete surfaces the behaviour of the liner is represented with an equivalent set of forces to simulate fully plasticized liner behaviour.
This model is used for high internal temperature conditions; severe accident, LOCA with design-basis earthquake.</td></tr>
<tr><td>Short-term</td></tr>
<tr><td>Thermal</td><td>Used for accidental thermal action</td></tr>
<tr><td rowspan="2">Model 2</td><td>Long-term</td><td rowspan="2">Represents all the concrete walls and includes the steel liner as an elastic membrane.
This model is used to study situations at the end of construction, in normal service, during tests, inspection earthquakes and design-basis earthquakes without thermal accident effects.</td></tr>
<tr><td>Short-term</td></tr>
</table>

701 Under the effects of accidental thermal loading in severe accident conditions, and under high temperatures and restraint from the concrete wall, the steel liner undergoes plastic deformation in all directions and exerts thrust on to the concrete. In this condition, ignoring work hardening, the liner cannot add any further stiffness or strength to the wall and so ignoring the liner as a resistant material, as in the case of Model 1, is appropriate.

702 In normal operating conditions and pressure test conditions, the steel liner behaves elastically and the unfavourable effect of liner thrust, where it is compressed through concrete pre-stressing, and concrete shrinkage and creep is taken into account in Model 2. This model is used for thermal load conditions where thermal gradients are associated with normal operation.

703 Thermal behaviour in the inner containment wall is determined by using a third model: Model 3 Represents the concrete walls and excludes the steel liner. The thermal inertia of the metal components (liner and penetration sleeves) is assumed to be zero.

704 In the mechanical analysis the concrete is represented using mostly 3D 20-node solid elements (ANSYS SOLID95) which are well suited to model curved boundaries. The FE mesh of the dome and wall uses three elements through the thickness, more at singular zones. Some degenerate forms such as prisms are used to zoom meshes at singular zones. The mesh is structured, appears to have been manually generated and very similar to the earlier COBEF model. Where modelled, the steel liner is represented using mostly quadrilateral 8-node curved shell elements (ANSYS SHELL93) which are well suited to model the containment geometry. Some triangular 6-node versions are used to

zoom meshes at singular zones. 96 2-node spring elements (assumed to be ANSYS COMBI165) are used to model simple linear springs to provide support stiffness.

705 For the thermal analysis similar order solid elements (ANSYS SOLID90) are used to the mechanical analysis, although formulated for steady-state and transient calculations, and with a similar mesh. As steel is assumed to have no thermal inertia, the liner and sleeves are not represented.

706 The dimensions of the elements are such that the maximum aspect ratio in the clear dome and wall parts of the wall is of the order of 8. Aspect ratios and the FE mesh is finer around singular zones.

707 Part of the outer containment wall (height of approximately 14.5m) is represented so that the bending stiffness of the outer containment wall, which acts to reduce gusset rotation, is included in the global FE model. The height required to produce full elastic interaction of the outer containment with the gusset is determined using simple shell theory.

708 To avoid directly representing the foundation soil in the global FE model a system of linear springs has been developed and equivalent stiffnesses calculated. In the global FE model three springs are arranged on each radial line of nodes at the base of the gusset for three degrees of freedom (vertical Kv, horizontal and radial Kh, and rotation in the radial direction Kr). Springs are attached to master nodes and the displacements of slave nodes controlled through displacement equations. They are located on the lower surface of the base of the model.

709 Assorted calculations and manipulations are undertaken to derive stiffnesses for inclusion in the different models. No evidence of verification associated with this overall methodology has been presented to ensure base reactions are consistent between the axisymmetric representation and the global 3D model. It is noted that from discussions with EDF and AREVA that the use of an axisymmetric model was an interim measure, that was necessary during design development for Flamanville 3.

710 In the case of the EPR at Flamanville, which is founded on granite, the approximations in the approach taken to the gusset support stiffness may not be significant as the foundation is near rigid. However, this approach would require detailed justification if applied to foundations in softer soil conditions. There is some limited evidence in Ref. EA184 that the influence of soft soil has been considered, although no further more detailed documentation has been referenced or made available.

711 The EDF and AREVA design basis specification NI hypothesis note Ref. EA82 describes the calculation of floor response spectra and earthquake displacements to be taken into consideration when designing items for use in standard buildings, including the inner containment wall, in the EPR plant series. Calculation note Ref. EA33 Section 4.3.10 lists the acceleration fields for the design-basis earthquake and indicates that these are obtained from the information detailed in hypothesis note Ref. EA82. However, whilst pseudo acceleration spectra are provided it is unclear how these are operated on to become static acceleration fields.

712 The methodology developed here by EDF and AREVA and applied by COB is not an established technique. This approach of applying an acceleration field calculated from the results of response spectrum analysis carried out using the global NI model appears to be inconsistent with The ETC-C. This concern has been reported by as part of RO-UKEPR-076 Action 3. Pseudo-static analysis and the use of acceleration fields to apply body forces in FE analysis is not established practice for the application of seismic loads in the analysis and design of critical structures. The principal concern would be the loss of

signal phase information. This has been captured as Assessment Finding **AF-UKEPR-CE-029.**

4.3.6.4.5.2 Inner Containment Detailed Design of the Liner

713 The inside of the inner containment wall is fitted with a steel liner, which is anchored to the concrete by stiffeners and connectors. Whilst the steel liner is not considered to be a 'structural component', it has an important function of maintaining leak tightness, especially in the event of an accident. It is also required to serve as formwork during construction.

714 Stiffeners and studded connectors are welded to the liner plate. The continuous stiffeners create a grid system which divides the liner into portions in accordance with geometry limits specified by the ETC-C Code. The liner is often in a compressive state, connectors were spaced in such a way that blistering is prevented or limited.

715 Ref. EA30 states that anchors must break before the liner to prevent cracking in the welds for leaktightness requirement. It also states that the liner thickness is restricted to prevent excessive thrust on the concrete containment wall, especially during thermal accidents.

716 Analyses were carried out to verify leak tightness and design compliance to the ETC-C Revision B. The steel liner with and without defects was assessed for loads during operation and in accidental cases. Analytical work was also undertaken to study liner behaviour during construction.

717 Different zones of the liner were modelled separately for areas with worst case loading and for major singularities. Finite Element Analysis (FEA) was carried out using SYSTUS and the analyses were elasto-plastic. Some specific zones were subject to analytical hand/manual calculations. These include the polar crane brackets, special plates of the equipment hatch and the smaller penetrations.

718 For all models with the exception of liner around openings (237mm to 1016mm diameter), the node coordinates and mesh information from the COB models were transferred to the relevant SYSTUS model. The boundary conditions for FE models were obtained from COB's modelling results, with strains from load cases applied to the anchors. The connectors were not represented in the FEA models. Springs were used to simulate the behaviour of the anchors embedded in the concrete wall. Imposed displacements were applied on the lines of the continuous anchors in the direction of the angles. The stiffness was assumed rigid (i.e. deformations are as per the input displacements) and in the perpendicular direction the stiffness was represented by spring elements, defined between the support node and the anchor beam element. The stiffness of the continuous anchorages was calibrated against a series of physical tests. For studies of operating conditions, the liner was fixed in the out-of-plane direction assuming restraint from the inner face of the concrete. Reactions were checked to ensure that if tensile forces are developed they remain small.

719 The elements used in SYSTUS have 5 layers which allow non-liner stress distribution through the thickness. Load effects were applied over a fictional time period to permit incremental non-linear behaviour. Von Mises stresses in the COB model were used to identify the most highly stressed region for various liner analyses.

720 Design criteria for the liners and anchors were based on the ETC-C Revision B, according to which there are no applicable criteria for the anchors in the 'defect-free' condition. However, the criteria for liner with defect was also used here. The worst case

analysis results were compared with the ETC-C Revision B limits and a margin calculated.

721 This is quite different to the work done for analysis of other parts of the liner. The approach to penetrations with diameters between 273mm and 1016mm were to analyse the extreme configurations with the largest and smallest openings. Unlike the other liner models, these models include the reinforcement annulus as well as plate elements for concrete. These areas were not modelled in detail for the COB concrete wall design and thus concrete and reinforcement are modelled in SYSTUS together with the steel liner to provide realistic boundary conditions.

722 The analysis of the steel liner with defects assessed two cases, liner buckling between connectors as a result of blistering and a failed ETC-C grillage. The worst case loading was obtained from defect free liner analysis for the cylindrical and dome sections. This analysis assumed an operational life of 60 years. Boundary conditions used are the same as for liner analysis without defects, with the stiffness of the anchors and connectors calibrated against a series of physical test results.

723 Characterisation tests for the connectors (Nelson studs) and stiffener angles were performed by EDF and AREVA in conditions which were thought to be as representative as possible. The EDF marginal analysis report Ref. EA227 summarises the most recent tests carried out and results obtained. The connectors were tested for shear and tension. The stiffeners were tested for shear, tension and bending.

724 The stiffener results were used to substantiate the compliance with ETC-C design criteria and were also used to define boundary conditions in the EPR liner design. For the connectors, the initial shear stiffness was used for the liner design. The connector tensile test result for the connector stem alone was used in design.

725 A different set of liner FE models were created for construction scenarios with specific boundary conditions set up for various phases of construction. Elementary load cases from climatic conditions (wind, snow and temperature) were considered in combination with pressure from concreting. The climatic conditions considered were specific to the Flamanville site. The concreting pressure was defined based on previous power plant constructions, with the maximum pour assumed to be 60cm high.

726 A series of singular zones have been examined in more detail, the gusset, equipment hatch, a typical smaller penetration and the polar crane supports. These are discussed in more detail below.

4.3.6.4.5.3 Inner Containment Detailed Analysis of Gusset

727 The gusset forms the connection between the inner containment wall and common raft and has to resist shear and bending from the containment principally due to pre-stressing, pressure during accident conditions and earthquake loading. It also has to react the horizontal loads from the reactor raft and reactor structures during earthquake loading. In this zone there is also the transition of the inner containment wall pre-stressed concrete, the passively reinforced concrete of the common raft and close by support to the passively reinforced outer wall.

728 A FE model has been developed of half the containment including the common foundation raft approximated to half a disk and including the lower parts of the inner and outer containment walls. The geometry of the containment internals is not represented. Instead, their impact on the behaviour of the containment wall base, particularly dead weight, is taken into account as an increased load or stiffness.

729 The interface between the containment and the raft is achieved through the use of springs applied to a master node and linked to slave nodes to provide a straight line boundary. In the case of FA3 founded on granite, the stiffnesses are so high that they have very little influence on the stress distribution in the gusset; the boundary is effectively rigid. For the UK EPR, when the foundation soil will be relatively soft,this approach is not seen as acceptable.

730 The primary reason for identifying this area for investigation of the application of ETC-C was that it was thought that the lower gusset may experience net tension in the concrete under some load cases with shrinkage. A typical zone of this interface is evaluated. As with the rest of the inner containment wall, the application of load cases and the requirements for reinforcement are almost entirely established using FE analysis and post-processing with in-house software. Ref. EA105 provides details of the reinforcement design for this area. This shear reinforcement, as with all other shear and longitudinal reinforcement in the inner containment wall, is calculated according to ETC-C using FERRAIL.

731 Document Ref. EA109 presents the loads and moment in the inner containment wall from the ANSYS analysis for each component of load, but does not show the combinations. It is difficult to read off accurate results from the graphs presented in Ref. EA109 however, there is a possibility that net hoop tension may exist in the gusset (or just above) under some combinations. As the combinations are not presented, we cannot conclude that the inner containment avoids net hoop tension under all load combinations. Discussion of this in technical meetings did not provide a satisfactory resolution. It is an assessment finding to investigate the potential for and resolution of the maximum net tensions in the gusset zone (Assessment Finding **AF-UKEPR-CE-40**).

732 During the review of the various analyses, a series of questions were raised under RO-UKEPR-076. The key areas where further information was required is as follows.

- Please demonstrate that the local stress conditions in the gusset singular zone (i.e. inner containment/basemat junction) have been considered and explain how the results have been interpreted for design purposes.
- The analysis methodology for the interaction between the inner containment model and global NI model presented in the documentation is insufficient in describing the detail of the process. Please provide a definitive statement as to how the boundary between the models has been represented and how the parameters have been determined.
- Equivalent static seismic analysis of the pre-stressed inner containment has been applied by using an acceleration field calculated from the results of response spectrum analysis carried out using the global NI model. Please provide details of this process.
- The stress and strain limits defined for Group 2 combinations in the ETC-C permit structural behaviour that may include cracking of the concrete and, for Group 3 combinations, may also include yielding of the reinforcement or liner. With reference to the ETC-C limits, please provide justification for the use of linear elastic analysis methods for the design of the inner containment.

733 The late delivery of the response to this RO has meant that its review cannot be included in the GDA Step 4 timeframe. Resolution of the RO-UKEPR-076 response concerns has been raised as GDA Issue **GI-UKEPR-CE-04**.

4.3.6.4.5.4 Inner Containment Detailed Analysis of Equipment Hatch

734 Sub-models have been developed for the analysis of the singular zones involving several over-lapping representations. For instance a CoB sub-model of the containment wall FE model has been developed around the equipment hatch to provide a finer mesh, which in turn is linked to separate NFM sub-models of the metal liner and the sleeve. The sub-models have been loaded using imposed displacements at their boundaries with pressure and temperature applied directly. In some cases imposed displacements are applied through a system of springs to represent liner stiffeners and always applied through a mesh transformation process. The interface between these models is complex and it is not clear how the imposed displacements derived from global models with approximate representation of singularity stiffness can be used as boundaries immediately adjacent to the more detailed FE models of the liner and sleeve without modifying the interacting load distribution. This is an assessment finding. The licensee shall justify the approach of using imposed displacements derived from global models with approximate representation of singularity stiffness as boundaries immediately adjacent to the more detailed FE models of the liner and sleeve without modifying the interacting load distribution (Assessment Finding **AF-UKEPR-CE-41**).

735 The review of this item has focused on the load cases and transfer of loads from the equipment hatch into the inner containment wall, with associated shear and bending reinforcement in the inner containment wall.

736 The structural design of the inner containment wall is substantially carried out using FE software and in-house post processing software. Supplementary further reinforcement is then added where necessary for example in singular areas such as around the equipment hatch.

737 In load cases where the liner yields (i.e. sudden increase in temperature) the results for the concrete are obtained by using a variant of the model (without the liner) and applying a set of nodal loads equivalent to those at which the liner yields. It is noted that this liner thrust load case is based upon the assumption that the liner is rigidly bonded to the concrete at all nodes.

738 A different approach is used for the equipment hatch which involves leaving the liner in place (elastic) then subtracting a uniform strain so that the mean hoop stress does not exceed the yield stress. Analysis is completed for all load combinations and reinforcement is automatically calculated to ETC-C by the software package FERRAIL.

739 The displacements from the ANSYS model are given to NFM for incorporation into their SYSTUS model for the liner design. It is noted that the interface presented here between the liner and the concrete is not a natural position for an interface between two design companies and does have risks associated with it. These include incompatibilities between the different models in terms of restraint/ loading conditions and displacements/ linearity of behaviour.

740 Finally hand calculations of reinforcement are prepared for specific areas such as the equipment hatch. The accidental load cases develop large shears exist around the equipment hatch. This shear reinforcement, as with most other shear and longitudinal reinforcement in the inner containment wall, is calculated according to ETC-C Revision B using FERRAIL. The design proceeds to use coloured contour plots (from FERRAIL) which are interpreted manually into bar diameters and spacing.

741 Under the accidental load cases the materials access sleeve must transfer large shear loads into the inner containment wall without splitting it. This is achieved via three rings on the sleeve which are cast into the inner containment. Rebar in the concrete is

included to allow the resulting shear transfer without splitting the wall. The process used to design this not contained in any philosophy documents, but a process was described verbally by EDF and AREVA. TQ-EPR-1431 was raised to seek clarification of the approach. This has not provided the necessary clarity. It is therefore an assessment finding for the licensee to provide detailed information and justification of the transfer of shear loads into the inner containment wall for penetrations (Assessment Finding **AF-UKEPR-CE-42**).

4.3.6.4.5.5 Inner Containment Detailed Analysis of Polar Crane Support

742 The design of the polar crane brackets has been based on a construction condition where no pre-stress tendons have been installed, where the dome is not built and when load testing is carried out. Separate checks are also carried out for the completed containment under normal operational conditions and under earthquake loading. These checks, focused on increased compression in the concrete, were carried out as part of the more general inner containment wall analysis. Testing of the polar crane takes place during construction before post-tensioning and before the dome is constructed. Consequently, the global FE model of the inner containment used in the design of reinforcement behind the crane brackets has been modified by removing the dome at level +46.5m.

743 The polar crane, is supported via four pairs of crane wheels. The wheels move along a circular crane runway beam which is in tern supported by 45 polar crane brackets which are anchored to the inner containment at equal intervals (every 8 degrees). Refs EA106 and EA110 provide further details on the polar crane bracket design methodology. It is noted that loading for the polar crane brackets is provided to the designers as loads on a single bracket only (i.e. not considering the case of adjacent brackets loaded).

744 The application of polar bracket loads to the FE model is by direct application of node loads. The application of the vertical load is associated with an eccentricity and, therefore, has an associated moment. Again this is applied directly to the relevant nodes. As solid elements are used to model the concrete wall, which have no rotational degree of freedom, the moment has to be applied to the shell element used to represent the steel liner. Internal actions have been determined by integrating stresses through the wall thickness using all eight actions to calculate reinforcement requirements with software FERRAIL.

745 This is an interface which places a possible limitation on the design as the methodology used as it appears to consider less shear than it would if a pair of loaded brackets were considered. Calculations for the loads provided to the designers, by the crane manufacturers, have not been seen. No radial inward load is considered in the design. This implies an assumption of zero transverse friction between the rails and the crane wheels, and is not obviously conservative. As a result, TQ-EPR- 1337 was raised. The response attempted to convey some complex concepts with limited information. The bogies which support the polar crane are self-standing with stabilisation wheels on both sides of the rail beam. The polar crane sits on the bogies via spherical bearings which act to reduce moment about an axis tangential to the crane rail, to a nominal value. Hence the support (without the radial wheels) is nearly a mechanism and only capable of resisting small lateral loads in a direction parallel to the crane bridge beam. To allow for the resistance of lateral loads a radial guide wheel is included in the design which is pre-loaded against the inside of the rail beam. This is clearly a complex arrangement, and whilst the response appears to be reasonable in principle, I remain to be fully convinced that assuming zero lateral load is valid. It is therefore an assessment finding that the

licensee shall justify all loads on the polar crane bracket, including the dismissal of radial inward loads (Assessment Finding **AF-UKEPR-CE-43**).

746 The design of the anchors is presented in Ref. EA108 using the methodology set out in Ref. OD37. The anchors are a mixture of headed studs, and deformed rebar of varying lengths and orientations arranged to avoid interaction with the tendon ducts. The methodology set out in Ref. OD37 does not include for distortion of the bracket (e.g. warping of the cross section) and does not consider the possibility that the different anchor types may have different stiffness values. Provided concrete pull-out is not a failure mode this is probably not significant at ULS. However, for the purpose of checking fatigue these aspects should be considered. TQ-EPR-703 was raised on this subject. The response is not entirely convincing.

747 The strength of the bracket is assessed analytically using hand calculations assuming the bracket, which is proportioned as a deep corbel, behaves as a beam where plane sections remain plane. Simple beam calculations are carried out and elastic buckling checks are carried out in accordance with Eurocode 3. Maximum Von Mises stresses are calculated and webs and flanges classified based on aspect ratios and checked against ULS code limits. It is considered that the assumption of plane sections remaining plane is not necessarily valid for this geometry.

748 Given the size and importance of the polar crane bracket we would expect to see some sort of verification with a Finite Element model which would include the parameters referred to above. The licensee will need to undertake a complementary analysis of a polar crane bracket to confirm that the assumed load and strain distribution within the bracket and the assorted anchorages is reasonable (Assessment Finding **AF-UKEPR-CE-44**).

749 The design of the anchorages has been undertaken using ETC-C, which defers to the CEB guide, although it modifies some parameters. A check of concrete cone failure for a group of anchorages is not presented nor is any assessment of anchor stiffness and bracket movement which is likely to be important at serviceability limits. This is considered a shortfall. Any licensee will need to undertake both group behaviour calculations as well as provide evidence that serviceability limits are complied with (Assessment Finding **AF-UKEPR-CE-45**).

750 The transfer of load from the polar crane bracket to the inner containment wall was designed by COB using a combination of strut and tie and FE. Strut-and-Tie is a design method for reinforced concrete members, especially suitable for the design of discontinuity regions (near support, in deep members, etc.), for which traditional beam equations do not apply. The strut-and-tie is aimed to transfer the load into the concrete local to the bracket. The FE model shows how the loads transfer more remotely from that area covered by the strut and tie model.

751 In the strut and tie methodology the anchor load transfer into the inner containment wall is designed only for a single loaded bracket and does not consider two adjacent anchors being loaded together. This could lead to an under prediction of shear which has to be transferred by the strut and tie system. The calculations should consider the effect of two adjacent brackets being loaded within the strut and tie system, or justification should be included in the calculations for not doing this. The consequences of any under-prediction of shear in the strut and tie model may be mitigated by the following points:

- the design load for one bracket is a large proportion of the total load for two wheels. Based upon this observation, it is felt that designing each bracket in isolation may not lead to problems with shear in the inner containment wall, and

- the FE analysis shows that the elastic stresses in the concrete one bracket away from the loaded bracket are relatively small compared to those in the immediate vicinity of the loaded bracket.

752 It is felt that this should be reviewed to quantitatively provide the justification for designing the strut and tie model for one loaded bracket in isolation. This is an assessment finding. At the site specific level, the licensee shall justify the strut and tie approach for the design of the containment wall adjacent to the polar crane brackets. The existing justification does not fully consider the distribution of loads in 3 dimensions (Assessment Finding **AF-UKEPR-CE-46**).

753 Fatigue design of the polar crane bracket or its anchors is not a requirement of ETC-C (Rev B). The AFCEN version of ETC-C has this requirement however. Fatigue has not been considered in the design of the polar crane bracket nor its anchors seen thus far. The response to TQ-EPR-736 suggests that this is not necessary but does not advance a fully convincing set of reasons. As part of the assessment finding to revalidate designs against the ETC-C AFCEN, this should be checked (Assessment Finding **AF-UKEPR-CE-07**).

4.3.6.4.5.6 Inner Containment Detailed Analysis of RPV Supports

754 The Reactor Pressure Vessel (RPV) is supported on a steel support frame which is in turn supported on a concrete support. The RPV support frame rests on the edge of the concrete pit allowing radial displacement and is held in place against horizontal rotational displacements due to horizontal loads such as the seismic loads by means of eight (8) equally spaced vertical keys incorporated in the lower plate and fitting into grooves in the foundation concrete. Radial movement of the RPV and attached piping (cold and hot legs) due to thermal expansion is allowed for in the design of the RPV support frame and in the design/interface of the RPV nozzle support with the RPV support frame. Each RPV nozzle is designed such that the lower portion of the nozzle is supported at each location by the RPV support frame. This interface prevents lateral movements from occurring (perpendicular to the piping), but allows for movement along the axis of the piping.

755 The review has examined Refs EA110, 112 and 113.

756 The distribution of lateral loads between the 8 keys is not clearly described or understood in Ref. EA110. It is also noted that the loads provided are indicated as “preliminary”, and that there is some inconsistency within Ref. EA110. It is therefore an assessment finding that the licensee shall confirm the loads imposed by the RPV ring on the civil structure and provide appropriate justification for their magnitude (Assessment Finding **AF-UKEPR-CE-47**).

757 The design of the lateral loads from the keys is seen as appropriate. The combined bending and shear case however has highlighted a number of concerns.

- Shear reinforcement is designed but extra reinforcement required to balance the shear compressive struts is not provided
- Shear links are shown without anchorage on each link the effective link areas assumed in the calculations would require the links to be bent over the longitudinal reinforcement at both ends.
- The horizontal beam is considered to be deep, but is analysed with bending theory rather than strut and tie.
- Transfer of loads between different sections of the support is not clear.

758 The conclusion from the above is that the detailed design of this area needs a much clearer explanation to provide confidence that the approach is correct. The licensee shall provide a revised set of calculations on the RPV support which provide a more complete justification of the approach adopted (Assessment Finding **AF-UKEPR-CE-48**).

4.3.6.4.5.7 Inner Containment Detailed Analysis of Tendon Anchorages

759 Horizontal pre-stressing tendons are anchored around the inner containment at three buttresses. Ref. EA174 provides details on the design approach and has been examined in some detail. The design is based on a mixture of European Technical Assessment guides (Ref. OD56), BSEN 1992-2 and guidance from Freyssinet.

760 The calculations have been found to be broadly acceptable, however there is a remnant concern over the choice of the primary regulation prisms and associated bursting reinforcement.

761 Some of the primary regularisation prisms are 1.456m long and BS EN 1992-2:2005 requires that anti-bursting reinforcement should be provided over the length of the prism. But the calculations only refer to anti-bursting reinforcement within 765mm of the face. Though it is noted that this latter dimension is in the manufacturer's literature. There seems to be an inconsistency between the primary regularisation prism chosen and the manufacturer's literature and French certification prior to ETA. At the stage of detailed site specific design, the licensee will need to confirm the design basis for the tendon anchorages and confirm adherence to the claimed codes and design standards (Assessment Finding **AF-UKEPR-CE-49**).

4.3.6.5 Beyond Design Basis Studies

762 As part of the response to RO-UKEPR-037, an examination of the beyond design basis performance of the containment was provided. This has not been found to be fully adequate in a number of areas (see Section 4.3.3.7).

763 A revised response to RO-UKEPR-037 has been received (Refs EA157 to EA164). This has not arrived in sufficient time to allow a meaningful assessment in the timeframe of GDA Step 4. Completion of this review is GDA Issue **GI-UKEPR-CE-03.**

4.3.6.6 Summary

764 The approach to the design of the containment is well understood, however there are key areas where EDF and AREVA have not provided sufficient information to allow a complete judgement to be made over the acceptability of their proposals.

765 The response to RO-UKEPR-037 on beyond design basis behaviour has been provided, but the assessment is still ongoing and is considered to be a GDA Issue (GDA Issue **GI-UKEPR-CE-03**).

766 The response to RO RO-UKEPR-76 on analysis of the containment is still ongoing and is considered to be a GDA Issue (GDA Issue **GI-UKEPR-CE-04**).

767 The approach of using grouted in place tendons is accepted in principle.

768 The monitoring and test systems are agreed in principle.

769 There are a large number of assessment findings.

770 The following Assessment Findings have emerged:

AF-UKEPR-CE-32: *The licensee shall develop a detailed grout specification for use at specific sites. This should be completed ahead of the installation of the Polar Crane. The sequence for grouting does not begin until after the Polar Crane has been installed. It is imperative that the grout specification is available before this stage in construction to allow sufficient time for trialling and mock ups.*

AF-UKEPR-CE-33: *The licensee shall develop a robust set of specifications for the grouting mock ups and the acceptance criteria. This should be completed ahead of the installation of the Polar Crane. The sequence for grouting does not begin until after the Polar Crane has been installed. It is imperative that the specification of the mock ups and the associated acceptance criteria are available to allow sufficient time ahead of actual grouting.*

AF-UKEPR-CE-34: *The licensee shall develop tensioning and grouting timing restrictions on a site specific basis. This should be completed ahead of the installation of the Polar Crane. The sequence for grouting does not begin until after the Polar Crane has been installed. It is imperative that the restrictions on timing are available before this stage in construction to allow sufficient time for the activities to be scheduled.*

AF-UKEPR-CE-35: *The licensee shall investigate the effects of bursting forces on the integrity of the containment wall and the effects of subsequent strain predictions (if any) incorporated into the proposed layout of containment instrumentation. This should be completed ahead of the placement of Nuclear Island safety-related. concrete. Modifications to the design may be necessary which could influence the tendon anchorages which are in the base of the NI slab.*

AF-UKEPR-CE-36: *The licensee shall develop the arrangements for the monitoring of the inner containment. This should be completed ahead of the placement of Nuclear Island safety-related concrete. Modifications to the design may be necessary which could influence the construction requirements.*

AF-UKEPR-CE-37: *The licensee shall develop the layout of the containment instruments sufficient to satisfy the requirements of the PCSR. This should be completed ahead of the placement of nuclear island safety-related concrete. Modifications to the design may be necessary which could influence the construction requirements.*

AF-UKEPR-CE-38: *The licensee shall develop the test criteria and related monitoring and alert arrangements for the initial and decennial pressure tests on the containment. This should be developed ahead of the containment pressure test.*

AF-UKEPR-CE-39: *The licensee shall investigate the potential for using Accoustic emissions and other non invasive detection systems on the inner containment structure as could be available to them at the time of site development. This should be developed ahead of the containment pressure test.*

AF-UKEPR-CE-40: *The licensee shall demonstrate the adequate treatment in the design of the Gusset area of any net tensions which develop**..** This should be completed ahead of the placement of nuclear island safety-related concrete. Modifications to the design may be necessary which could influence the construction requirements.*

AF-UKEPR-CE-41: *The licensee shall justify the approach of using imposed displacements derived from global models with approximate representation of* *singularity stiffness as boundaries immediately adjacent to the more detailed FE models of the liner and sleeve without modifying the interacting load distribution. This should be completed ahead of the placement of nuclear island safety-related concrete. Modifications to the design may be necessary which could influence the construction requirements.*

AF-UKEPR-CE-42: *The licensee shall provide detailed information and justification of the transfer of shear loads into the inner containment wall for penetrations. This should be completed ahead of the placement of nuclear island safety-related concrete. Modifications to the design may be necessary which could influence the construction requirements.*

AF-UKEPR-CE-43: *The licensee shall justify all loads on the polar crane bracket, including the dismissal of radial inward loads. This should be completed ahead of the placement of Nuclear Island safety-related concrete. Modifications to the design may be necessary which could influence the construction requirements.*

AF-UKEPR-CE-44: *The licensee shall undertake a complementary analysis of a polar crane bracket to confirm that the assumed load and strain distribution is reasonable. This should be completed ahead of the placement of Nuclear Island safety-related concrete. Modifications to the design may be necessary which could influence the construction requirements.*

AF-UKEPR-CE-45: *The licensee shall undertake both group behaviour calculations as well as provide evidence that serviceability limits are complied with for the Polar Crane anchorages. This should be completed ahead of the placement of Nuclear Island safety-related concrete. Modifications to the design may be necessary which could influence the construction requirements.*

AF-UKEPR-CE-46: *The licensee shall justify the strut and tie approach for the design of the containment wall adjacent to the polar crane brackets. The existing justification does not fully consider the distribution of loads in 3 dimensions. This should be completed ahead of the placement of nuclear island safety-related concrete. Modifications to the design may be necessary which could influence the construction requirements.*

AF-UKEPR-CE-47: *The licensee shall confirm the loads imposed by the RPV ring on the civil structure and provide appropriate justification for their magnitude. This should be completed ahead of the placement of nuclear island safety-related concrete. Modifications to the design may be necessary which could influence the construction requirements.*

AF-UKEPR-CE-48: *The licensee shall provide a revised set of calculations on the RPV support which provide a more complete justification of the approach adopted. This should be completed ahead of the placement of Nuclear Island safety-related concrete. Modifications to the design may be necessary which could influence the construction requirements.*

AF-UKEPR-CE-49: *The licensee shall confirm the design basis for the tendon anchorages and confirm adherence to the claimed codes and design standards. This should be completed ahead of the placement of Nuclear Island safety-related concrete. Modifications to the design may be necessary which could influence the construction requirements.*

AF-UKEPR-CE-68: *The licensee shall undertake analysis of the containment structure to reflect the actual concrete properties used in the construction. This* *shall be undertaken ahead of the containment pressure test to provide a credible comparison for the measured values.*

4.3.6.7 Aircraft Protection Structures

4.3.6.7.1 Scope

771 Aircraft protection structures are provided for elements of the Nuclear Island and for the cooling water pumphouse. The design of the pumphouse is not considered as part of the GDA as it is a site specific structure.

772 The Nuclear Island aircraft shell is designed to protect the Reactor Building, Fuel Building and Divisions 2 and 3 of the Safeguard Building against military and commercial aircraft crashes. It takes the physical shape of a thick wall which covers the roofs, and surrounds the outer walls of the Fuel Building and Divisions 2 and 3 of the Safeguard Building. The outer containment also provides the same protection at its dome and at the vertical upper section facing divisions of Safeguard Buildings 1 and 4. Additionally the vertical outer walls of the staircases for personnel access to the Nuclear Island buildings form columns which are part of the aircraft shell.

4.3.6.7.2 Standards

773 Aircraft impact is considered under the aegis of external hazards. For accidental aircraft impact it is possible to calculate some frequency relationship between the likelihood of impact and the nature of the aircraft. For malicious impact this is not practicable and a deterministic approach is required. The nature of the malicious threat is not discussed further in this report.

774 Malicious aircraft crash is considered as a beyond design basis accident. The guidance within the SAPs (FA.15 and FA.16) will be used as guidance. In addition, guidance from Ref. OD10 will also be considered.

4.3.6.7.3 Assessment

775 During Step 2, EDF and AREVA provided clarification on their position with respect to commercial airliner impact, which is repeated below;

- The systems important for the safe operation of the reactor are protected against aircraft impact either by a thick concrete shell or by physical separation (duplicated systems are located in separate areas which could not be affected simultaneously by a single aircraft impact).
- The original design basis of the plant took into consideration indirect and potential consequences of aircraft impacts for the cases of general aviation and military aircraft. After the events of September 11th 2001 the design was verified and modified as necessary to address the possibility of the direct impact of a large commercial airliner.

776 The finite element codes used have been reviewed at a principles level and found to be adequate for the purposes they have been used for. The design codes used have been examined through dialogue with the code committee members and a review of relevant supporting documentation. In addition, it should be noted that their use has been previously sanctioned by ASN STUK and the German federal regulator BMU.

777 The loading functions and scenarios associated with military and commercial (accidental and non-accidental) aircraft impact have been examined and found to be adequate.

778 The analyses undertaken for military and commercial impact have been examined and found to be adequate to predict the loads and displacements within the structures.

779 The key claims on the ability of the aircraft protection shell to provide sufficient protection against the loss of key safety functions has been found to be satisfied.

780 The detailed assessment of the capability of the doors and openings within the APC shell has not been undertaken. A review of the principles to be employed for the design of doors and openings has been undertaken, and has been found to be broadly acceptable. There remains an assessment finding for the licensee to provide detailed justification of the design of APC shell doors and openings (Assessment Finding **AF-UKEPR-CE-67**).

781 The overarching safety justification has been assessed and found to be broadly acceptable with a small number of modifications required which can be addressed on a site specific basis or can be addressed generically.

782 There will still be a requirement to undertake site specific analyses of the behaviour of the Nuclear Island under aircraft impact to ensure that the design envelope of induced vibrations for safety critical plant and equipment is not breached. The licensee shall undertake site specific analyses of the behaviour of the Nuclear Island under aircraft impact to confirm the in-structure responses are within the GDA envelope (Assessment Finding **AF-UKEPR-CE-50**).

783 The probabilistic study of accidental aircraft impact will need to be examined in more detail on a site specific basis. The licensee shall undertake a probabilistic study of accidental aircraft impact on a site specific basis (Assessment Finding **AF-UKEPR-CE-51**).

4.3.6.7.4 Summary

784 The APC shell for the Nuclear Island is considered to provide sufficient protection against commercial and military aircraft impact.

785 The approach to protection of the EPR in general against aircraft impact has been found to be satisfactory, with a small number of assessment findings to be undertaken on a site specific basis as listed below:

> ***AF-UKEPR-CE-50:*** *The licensee shall undertake site specific analyses of the behaviour of the Nuclear Island under aircraft impact to confirm the in-structure responses are within the GDA envelope. This should be completed ahead of the installation of the Polar Crane. The rationale for this is that there may be a requirement to modify the qualification of the crane control equipment as a result of this analysis.*

> ***AF-UKEPR-CE-51:*** *The licensee shall undertake a probabilistic study of accidental aircraft impact on a site specific basis. This should be completed ahead of the installation of the Polar Crane. The rationale for this is that there may be a requirement to modify the qualification of the crane control equipment as a result of this analysis.*

> ***AF-UKEPR-CE-67:*** *The licensee shall provide detailed justification of the design of APC shell doors and openings. This should be completed ahead of the placement of Nuclear Island safety-related concrete. Modifications to the design may be necessary which could influence the construction requirements.*

4.3.7 Ancillary Structures

786 Section 4.2 contains an overview of the coverage of the GDA in terms of structures being considered as 'generic' and those which can only be reviewed in detail once the site specific designs have been undertaken.

787 Nonetheless, the classification and overall claims made on the structures can be reviewed at this stage, even if the detailed design cannot be examined.

4.3.7.1 Scope

788 The following structures are discussed in this section

- Waste Treatment Building.
- CW Pumphouse.
- Diesel Buildings.
- Ancillary Buildings.
- Ancillary services and structures, i.e. tanks, service trenches .
- Turbine Hall.

4.3.7.2 Standards

789 There are no particular standards which apply to this section, rather it is a statement of what we have been able to undertake within GDA, and an identification of these aspects which will need to be considered in site licensing.

4.3.7.3 Assessment

790 The following sections provide a summary of the findings on individual structures.

4.3.7.3.1 Assessment: Waste Treatment Building

791 The waste treatment building for the UK application has not been designed as yet. There has been no presentation of any kind of hypothesis note either. This is out of the scope of GDA.

4.3.7.3.2 Assessment: CW Pumphouse

792 The CW pumphouse has been designed in detail for the Flamanville and Olkiluoto sites. The intake arrangements there are a canal type system, which may not be representative of UK sites, where an intake tunnel and forebay arrangement is more likely. Additionally, due to the semi – embedded nature of the structure, the site specific soil characteristics will dictate the nature of the detailed design of the ECCS. The structure is classified as safety class C1 and seismic class SC1 and Trains 1 and 4 are protected against aircraft impact. The classification of the structures is considered to be appropriate, however no further assessment has been undertaken during GDA. There has been no presentation of any kind of hypothesis note either. Assessment of the CW pumphouse building will be a site specific activity and is out of the scope of GDA.

4.3.7.3.3 Assessment: Diesel Buildings

793 The diesel houses have been designed in detail for the Flamanville and Olkiluoto sites. Their safety classification appears to be reasonable. They have been declared out of scope for the GDA however.

794 Prior to being removed from the scope, some examination of the documents EA48,49 and 50 had been undertaken. They currently do not contain sufficient information to allow them to be used in the UK as hypothesis notes for the design of the Diesel Building. This is part of GDA Issue **GI-UKEPR-CE-01**.

4.3.7.3.4 Assessment: Ancillary Buildings

795 The design of other ancillary structures which have a safety role is out of the scope of GDA and will be a site specific activity.

4.3.7.3.5 Assessment: Ancillary Services and Structures, i.e. Tanks, Service Trenches

796 The detailed design of the service trenches is out of the scope of GDA and will be a site specific activity.

4.3.7.3.6 Assessment: Turbine Hall.

797 The Turbine Hall is not a safety classified structure, however it is seismic class SC2, but has no aircraft protection.

798 Ref. EA72, the Flamanville 3 hypothesis note for the turbine hall has been reviewed and the following comments made.

- The seismic design criteria needs expanding considerably, especially the way that the defined demand is integrated into EN1998 and the use of the UK national annexe.
- There are no statements over detailed design requirements for drop loads or turbine disintegration considerations.
- The sections on Climatic loads will need updating for a UK context.
- The guidance on the construction of the finite element models for the structure is very weak without reference to other guidance.
- There is no reference to the need to consider decree 93-1418 or 94-1159, the French equivalent of the CDM regulations.

799 There is a need for the Turbine Hall hypothesis note to be upgraded to include consideration of the comments raised in this report. The licensee shall develop a hypothesis note for the Turbine Hall (Assessment Finding **AF-UKEPR-CE-52**).

4.3.7.3.7 Findings: Main Control Room

800 The Main Control Room (MCR), as its name suggests, houses the main control systems, equipment and consoles of the UK EPR plant. It is also where all operating data are centralised. It is located within the Safeguard Auxiliary Building and is permanently occupied.

801 Refs EA47, EA238 and EA239 have been reviewed as part of the assessment.

802 The concept adopted for the MCR is a so-called "Box-in-Box" type of construction. A welded structural steel "box" of steel columns, girders and beams is inserted into the concrete "box" of slabs and walls of the SAB. The structural steel "box" is supported on 18 spring-damper elements which are in turn supported on the concrete floor slab of SAB 2. The 18 spring-damper elements are arranged in 3 rows of 6 elements. These spring damper elements are referred to as GERB springs (www.gerb.com).

803 The primary objective is for noise and vibration isolation for achieving a user friendly environment for human factors. The EDF technical specification report (Ref. EA47) has set a limit on the structural borne noise level at 45 dBA.

804 The secondary objective and therefore a by-product of this concept is to use the 18 spring-damper elements as both vertical and horizontal seismic isolation bearings/isolators, such that both the vertical and the horizontal motions of the concrete floor slab of the SAB 2 induced by the DBE are substantially reduced when transmitted from the concrete floor slab across the spring-damper elements to the structural steel "box" which houses the MCR.

805 The benefit of seismic isolation is the reduced vertical and horizontal floor response spectra demands for seismic qualification of the components and equipment within the MCR.

806 The adopted "box-in-box" design concept is also called "floating floor" or "room-in-room" structures for effectively reducing transmission of vibration and structural borne noise within buildings. This design concept is often used to decouple sensitive performance centres like TV, broadcasting or recording studios, recital and rehearsal rooms, theatres as well as gymnasia from the surrounding structure.

807 The use of soft spring elements, with or without damper elements, for noise and vibration reduction is a mature technology widely used for mitigation of noise and vibration induced by machine operations, rail and road traffic and so on. Such spring elements or combined spring-damper elements isolate the vertical vibration. These mechanical devices can also isolate the vertical motions induced by a seismic event and thereby act as vertical seismic isolation devices.

808 One main consideration of the practicality of this concept is to provide in the design for access and space for possible replacement of every single one of the installed spring-damper elements. The Sofinel technical report did not outline a strategy for possible replacement of the spring-damper units. During discussions with EDF and AREVA it became clear that the detailed design of the steel frame has allowed for the need to remove and replace each of the GERB springs.

809 A SOFiSTiK model of the MCR (Refs EA238 and 239) has been developed including discrete representation of each of the spring damper elements. A review of this model has found it to be adequate, however there were some minor concerns over the modelling of damping. These were resolved through TQ-EPR-1132 in terms of the response of the structure itself. However, the derivation of in-structure spectra for qualification of equipment will need to consider this in more detail, however this is out of the scope of GDA.

810 The predicted movements of the structure on the springs are below the limits the manufacturer places on them, with typically on 75% of allowable movement being used.

811 Ref. EA47 Section 3 "Design Criteria" outlines a methodology for checking the internal forces of structural steel members. Equation (3.2) is merely an equation to calculate the normal stress caused by the axial force N and a bending moment M_y at a cross section of

beam-column members. This calculated stress is then compared against the design yielding strength f_{yk} = 235 MPa. This methodology for checking structural steel members is not acceptable. This must be carried out in accordance with an accepted structural steel building design code. The use of the methods in the ETC-C would be acceptable. It is an assessment finding that the design of the steel box of the MCR should be in accordance with the ETC-C (Assessment Finding **AF-UKEPR-CE-53**).

812 The analysis of structure and airbourne noise into the MCR has only been reviewed briefly. Letter EPR00778 provided some further information justifying the approach. This is being examined as part of the Human Factors topic area and is not considered further here.

813 Within the response to Letter EPR00778, clarification was provided on the classification of plant and equipment and sub-structures within the MCR. The responses are generally seen as appropriate, however it is not clear why the cable trays are seismic class 2 if they are carrying essential signals. During the site specific design, justification of the seismic class of all items of structures systems and components in the MCR, will be required (Assessment Finding **AF-UKEPR-CE-54**).

4.3.7.3.8 Assessment of Nuclear Auxiliaries Building (NAB) Chimney

814 The NAB chimney hypothesis note (Ref. EA41) was reviewed and a series of comments provided to EDF and AREVA under TQ-EPR-1185. The response to this TQ was broadly acceptable; EDF and AREVA agreed to re-issue the hypothesis note for a UK EPR (Assessment Finding **AF-UKEPR-CE-55**).

4.3.7.4 Summary

815 The bulk of the structures which do not form the Nuclear Island have not been examined in detail.

816 For those structures that have been examined it is clear that the hypothesis notes are inadequate in their current form, and a series of assessment findings have been generated.

> ***AF-UKEPR-CE-52:*** *The licensee shall develop a hypothesis note for the Turbine Hall. This should be completed ahead of the placement of first structural concrete. Modifications to the design may be necessary which could influence the construction requirements.*

> ***AF-UKEPR-CE-53:*** *The licensee shall ensure that the design of the steel framework of the MCR is undertaken in accordance with the ETC-C AFCEN. This should be completed ahead of the installation of the Polar Crane. The timing of this activity is not correlated to the installation of the polar crane, however the timing of the installation of the MCR is broadly compatible with crane installation.*

> ***AF-UKEPR-CE-54:*** *The licensee shall provide justification of the seismic class of all items of structures systems and components in the MCR. This should be completed ahead of delivery to Site of Mechanical. Electrical and C&I Safety Systems.*

> ***AF-UKEPR-CE-55:*** *The licensee shall produce a revised version of the NAB Chimney hypothesis note for the UK EPR. This should be completed ahead of the placement of first structural concrete. Modifications to the design may be necessary which could influence the construction requirements.*

4.3.8 Beyond Design Basis

817 PCSR Sub-chapter 3.3 Section 1.3.5 talks about the inclusion of margins in the design of the EPR civil structures.

Sub-chapter 3.3 Section 1.3.5 states that:

> *"External events, where the design of the structures must make provision for loadings, whether they are due to natural phenomena (i.e. earthquakes or climate change) or human induced events (e.g. explosions or aircraft crash"*

818 There has been little evidence presented of a coordinated attempt to include this provision within the design of key structures. As part of the site specific design, the licensee will need to provide evidence that the design has incorporated beyond design basis considerations. This is an assessment finding:

> ***AF-UKEPR-CE-66:*** *The licensee shall demonstrate that adequate margins beyond the design basis exist for all Class 1 civil structures. This should be completed ahead of the placement of Nuclear Island safety-related concrete, as the potential exists for the design to be modified as a result of this review.*

4.3.9 Decommissioning

819 At the end of the facilities operating life, there will be a need to decommission. Decisions made and design choices made at this stage can significantly affect the ease of decommissioning. It is therefore important that the dismantling of the structures is examined in more detail at this stage.

820 This report focuses solely on the decommissioning of the civil structures, and does not directly address the issues of decontamination or treatment of radioactive waste.

4.3.9.1 Scope

821 This report focuses solely on the decommissioning of the civil structures; Nuclear Island and ancillary structures. It does not discuss any structures developed as part of construction such as marine offload facilities or transport links. In addition, any structures required for the provision of cooling water or protection against flooding or other natural hazards which are developed as part of site specific activities are out with this scope.

4.3.9.2 Standards

822 The key SAPs which are applicable to this area are as follows.

Decommissioning	Design and operation	DC.1
Facilities should be designed and operated so that they can be safely decommissioned.		

> *Account should be taken during the planning, design, construction and operational stages of the need for decommissioning and waste retrieval. This should include:*

a) design measures to minimise activation and contamination, etc;
b) physical and procedural methods to prevent the spread of contamination;
c) control of activation;
d) design features to facilitate decommissioning and to reduce dose uptake by decommissioning workers;
e) consideration of the implications for decommissioning when modifications to and experiments on the facility are proposed;
f) identification of reasonably practicable changes to the facility to facilitate or accelerate decommissioning;
g) Minimising the generation of radioactive waste.

4.3.9.3 Assessment

823 Chapter 16 of the PCSR addresses decommissioning. The overall declared strategy for structures is as follows. "*In the current strategy the end state (brownfield) is assumed to include the radiological decontamination of all buildings and their demolition to one metre below ground level, then backfill and grading of voids*." (see Ref. EA250).

824 The structures used in the EPR are heavily reinforced concrete and pre-stressed concrete structures, and do not really have any significantly novel features in their design or construction which requires novel solutions for their demolition. In this regard they can be seen to be very similar to heavy bridge or other industrial structures.

825 The proposed approach for demolition using a combination of hydraulic power shovels, diamond wire cutting, long arm hydraulic rock breakers is seen as standard practice, and is seen as entirely practicable.

826 For the inner containment, there is considerable contained energy stored in the pre-stressing tendons, and this needs to be considered carefully before deciding on the approach to be adopted. The initial suggestions from EDF and AREVA in the response to TQ-EPR-513 are that a mixed approach of mechanical demolition and use of explosives would be adopted. This approach is entirely consistent with current practice for large pre-stressed structures such as bridges and liquid natural gas tanks.

827 The decommissioning of the fuel pool structures may be a slightly more involved process, due to the potential for contamination in the leak detection channels and pipework embedded in the reinforced concrete walls. There are no fundamental concerns over the practicability of this however.

828 In some areas of the plant, specific design requirements are placed to limit the crack size in concrete to ensure that if there is a spillage, there is limited ingress of potentially contaminated liquid into the structure.

4.3.9.4 Summary

829 The decommissioning of the EPR structures is considered to be achievable with current demolition techniques, although a carefully developed strategy for the inner containment structure will be required.

4.3.10 External Hazards in Design

830 The identification, screening and consolidation of the external hazards into loadings for plant and equipment has been discussed elsewhere in this report. This section focuses on the incorporation of this information into the safety justifications for the plant.

4.3.10.1 Fault Schedule

831 The fault schedule thus far presented does not include any specific consideration of external hazards as initiating events. RO-UKEPR-43 raised this as "*Action 4: Further clarity on the consideration of external hazards in the Fault Schedule is required".* The response thus far has been disappointing. It is therefore an assessment finding that the licensee shall develop a fault schedule incorporating external hazards (Assessment Finding **AF-UKEPR-CE-56**). This should be completed ahead of the placement of first structural concrete. Modifications to the design may be necessary which could influence the construction requirements.

4.3.10.2 Consolidated Safety Case

832 The claims made against external hazards within the PCSR are at a high level. They do not constitute what would be considered a detailed safety case against external hazards. This is not unexpected at a generic level, as there are clearly a number of hazards which cannot be clearly defined until a site is chosen. It is therefore an assessment finding that the licensee shall develop a consolidated external hazards safety case (Assessment Finding **AF-UKEPR-CE-57**). This should be completed ahead of Fuel Load.

4.3.10.3 PSA Modelling

833 The PSA models will incorporate external hazards as key drivers. There is a need to review the base data provided into the models to ensure its validity. The review of the PSA is reported elsewhere.

834 The key Steps in the assessment process have been:

- Establish key importance items from PSA.
- Review fragility curves produced for high importance items.
- Review logic tree for high importance legs for consistency with design intent.

835 Refs TSC41 and TSC42 provide further details on the assessments undertaken.

836 EDF and AREVA have undertaken Probabilistic Safety Analyses (PSA) for the UK EPR, as described in Chapter 15 of the PCSR (Ref. EA1). As listed in Section 1 of Sub-chapter 15.0, probabilistic considerations are used for a variety of purposes, including:

- *"To calculate the plant seismic capacity in order to demonstrate that the plant has sufficient margin beyond the safe shutdown earthquake (see Sub-chapter 15.6)"*

With regard to external hazards, Sub-section 2.2 of Sub-chapter 15.0 states:

- *"PSA Safety Objective 3: As a general rule, design measures must be taken for external hazards consistent with those taken for internal events and hazards: Thus, the external hazards should not make up a large part of the overall core damage risk (Paragraph A.2.5 of the Technical Guidelines – Sub-chapter 3.1 Table 1)."*

837 Based on this, it would be expected that a full seismic PSA be carried out. However, it is evident that only a seismic margins assessment has been performed. Sub-chapter 15.6 of the PCSR presents the Seismic Margin Assessment (SMA). Section 1 states:

- *"NII Safety Assessment Principles and EUR design principles require an additional demonstration that the reactor design is robust against events more severe than that assumed for the plant design, so that no 'cliff edges' exist beyond the design basis. The purpose of the current Seismic Margin Assessment (SMA) is to demonstrate that this requirement is achieved so that safe shutdown can be achieved in seismic events that exceed the DBE by a certain amount"*
- *The seismic margin of the UK EPR is assessed by a PSA-based SMA, following a methodology developed by the US NRC. This approach uses the PSA model to identify combinations of seismic equipment failures which could result in core damage, as well as combinations of seismic failures, random failures and human errors which contribute significantly to seismic risk. By identifying which equipment items and structures are of critical importance in seismic events, the analysis approach ensures that vulnerabilities in the design are identified allowing them to be corrected if necessary, thus helping ensure that the seismic risk is ALARP"*

838 There is no commitment to perform a seismic PSA beyond the GDA process, and hence the seismic margin work is the only basis on which acceptability is demonstrated.

839 Seismic margins are developed for various success paths, guided by the PSA, and are presented in terms of High Confidence of Low Probability of Failure (HCLPF) values. EDF and AREVA have used a target level Seismic Margin Earthquake (SME) of 1.6 times the 0.25g pga design basis earthquake (DBE) as the basis of their seismic margins work, with the aim of demonstrating a success path with a HCLPF value in excess of 0.4g pga.

840 The PCSR does not specifically list the supporting fragility calculations. These calculations are presented in Refs EA232 and EA233. It should be noted that there are a number of items remaining open in these documents. These items concern assumptions made regarding anchorage design / governing failure mode, anchorage capacity and the necessity to review results when site-specific information becomes available.

841 Ref. EA232 presents the following calculations:

- **Building Fragilities**
  - Inner containment structure
  - Internal structure
  - Safeguard Buildings 2 and 3
- **Mechanical Component and System Fragilities**
  - Pumps
  - Tanks
  - Heat exchangers
- **Distribution System Fragilities**
  - Piping
  - Valves
  - Cable Raceways

- **Active Electro-Mechanical Component Fragilities**
    - MCC and switchgears
    - I&C panels
    - Inverters and battery chargers
    - Batteries
    - Transformers
- Ref. EA233 presents the following calculations:
    - Primary Equipment Fragilities
    - Fuel grid
    - Reactor core internals
    - Reactor pressure vessel (RPV)
    - Steam generator
    - Reactor coolant pump
    - Pressuriser
    - Control rod drive mechanism (CRDM)

842 Once a fragility has been calculated it is assigned to one or more appropriate equipment categories e.g. the fragility calculated for ‘Electrical equipment qualified to HN 20-E-53’ is assigned to the appropriate equipment categories that include ‘battery’, ‘charger’, ‘transformer’ etc. After all the equipment categories have been assigned a fragility, the Seismic Equipment List (SEL) can be populated e.g. the SEL item ‘Batteries & Racks (220V DC)’ is assigned the fragility of the ‘battery’ equipment category.

843 A similar process is used for structures e.g. the fragility calculated for ‘Internal structure’ is assigned to appropriate SEL items such as ‘Fuel Transfer Tube’, ‘Core Melt Retention Structure’ etc. The SEL, populated with the calculated fragilities, is presented in both the PCSR and Ref. EA232

844 A representative selection of calculations was chosen from Ref. EA232 and the items with the lowest HCLPFs were identified from Ref. EA233. On this basis, the list of chosen items was as follows:

- Building Fragilities
    - Inner containment structure (rock site).
    - Internal structure (rock and soil site).
    - Safeguard buildings 2 and 3 (soil site).
- Mechanical Component and System Fragilities
    - Pumps including anchorage (rock site).
- Active Electro-Mechanical Component Fragilities
    - Electrical equipment qualified to HN 20-E-53 including anchorage (rock site).
- Primary Equipment Fragilities
    - Fuel grid EOL 7 in row (soil site).

- o Reactor core internals (rock site).
- o Steam generator snubbers (rock site).
- o CRDM (rock and soil site).

845 The fragility calculations have made use of as much as possible of seismic design and qualification information. It is noted by EDF and AREVA that this is an intermediate step in the PSA-based SMA. Currently assumptions have been made and stated due to limited plant-specific information and no site-specific data.

846 The SMA is based on the following:

- Plant-specific data (this is historical data, and excludes UK EPR site licence application plant-specific data)
  1. Design analysis
  2. Test data
- Generic data
  1. Earthquake experience data
  2. Fragility test data
  3. Qualification tests of similar components in other plants
  4. Expert opinion
- Fragility parameter values derived from past PRA

847 The first step in the SMA is to define a review level earthquake, typically named a Seismic Margin Earthquake, as identified in Ref. OD40.

848 The UK EPR will be designed to withstand the EUR spectrum (Ref. OD52) anchored to 0.25g pga as the DBE using the codes ETC-C, RCC-M and ASME.

More specifically, the seismic design parameters for GDA are as follows:

- EUR anchored to 0.25g pga. These include EURS, EURM and EURH depending on site condition.
- Vertical spectrum is $^{2}/_{3}$ of the horizontal.
- Soil SA, MA, MB, MC, HA and HF are to be considered.

The plant will be designed for the envelope of these. Note that Flamanville layered hard soil condition was included in this envelope.

The target SME for UK EPR has been selected by EDF and AREVA to be defined by 0.4g pga and anchoring the appropriate EUR spectrum for the site where the plant will be located (EURS, EURM, EURH), thereby defining the SME.

849 In the fragility calculations, two UK sites (Site 2 – medium site 0.27g pga and Site 4 – rock site 0.19g pga) for which site-specific UHS at 10^{-4} are available have been considered by EDF and AREVA.

850 The methodology that has been utilised by EDF and AREVA to determine the HCLPF values is the probabilistic fragility analysis method. The guidance in Ref. OD49 and Ref. OD50 is used extensively in the seismic margin calculations. In addition, EPRI have produced a Seismic Fragility Application Guide Ref. OD51. Ref. OD51 supplements the

data in Ref. OD50 with further worked examples, but does not alter the approach to their derivation.

851 The approaches used in these documents represent best practice in this field. They have been extensively reviewed by the US NRC. The approach is therefore considered acceptable in principle.

852 Ref. TSC42 has undertaken a detailed review of the fragility calculations. A summary of the results of this review can be seen in Table 9 below.

Table 9: Summary of Seismic Fragilities Reviewed

Item and Failure Mode Assessed	Method of Assessment	HCLPF (g pga)	Key Results of Review
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted] • [redacted] • [redacted] [redacted] • [redacted] [redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted] • [redacted]

Item and Failure Mode Assessed	Method of Assessment	HCLPF (g pga)	Key Results of Review
			[redacted] • [redacted] • [redacted] [redacted] • [redacted] [redacted] • [redacted]
[redacted]	[redacted]	[redacted]	[redacted] • [redacted] • [redacted] [redacted]

853 In summary, the approach used has been found to be broadly acceptable. It should be noted however that there will be a need to re-evaluate these on a site specific basis. There are a number of other aspects that will need to be examined on a site specific basis.

- Relay chatter has been screened-out because it is stated that solid state relays are to be used. These were stated to be inherently immune to mechanical switching discontinuities and hence need not be considered in the SMA. This is judged acceptable. However, so called solid state relays can sometimes still include printed circuit board mounted relays as a final output stage. The licensee shall confirm that relay chatter is not a concern for the proposed plant and equipment for a particular site either through elimination of components which exhibit this behaviour or by suitable testing (Assessment Finding **AF-UKEPR-CE-**

58).

- The site-specific horizontal UHS, based on or corrected to be the maximum of two horizontal directions (as opposed to the geometric mean), must be shown to be enveloped by the 0.25g pga design spectrum and the two UK sites (Site 2 – medium site 0.27g pga and Site 4 – rock site 0.19g pga) for which site-specific uniform risk spectra at 10^{-4} per annum which were used by EDF and AREVA.
- The site-specific vertical UHS must be shown to be enveloped by the 0.167g pga design spectrum and the two UK sites (Site 2 – medium site 0.27g pga and Site 4 – rock site 0.19g pga) for which site-specific uniform risk spectra at 10^{-4} per annum which were used by EDF and AREVA. The licensee shall confirm that the seismic fragilities used are valid for the particular site conditions (Assessment Finding **AF-UKEPR-CE-59**).
- For electrical equipment to be qualified by shake table testing, no factor for variability has been included for the boundary conditions for the future installation of the equipment being different from the boundary conditions that will be used for the testing of equipment. The implicit assumption is that the site installations will be consistent with those adopted for the testing.

854 In addition to consideration of seismic hazards, the fragility of the containment against overpressurisation has been estimated for use within the Level 2 PSA models. The Level 2 PSA study identifies, evaluates and quantifies loads on the containment structure that can occur as a result of a severe accident. In order to assess the probability that a given load will result in failure of the containment structure, knowledge of the capacity of the structure to withstand severe loading is required. EDF/AREVA have performed a fragility evaluation of the containment to use in the Level 2 PSA, and it is this containment fragility evaluation which has been assessed as part of GDA. The PCSR presents the fragility results for six potential over-pressurisation failure modes for the containment, two being associated with equipment hatches.

855 The list of chosen items for detailed review was as follows:

- Typical zone in the containment cylindrical wall.
- Singular zone at the base of the containment cylindrical wall.

856 Ref. EA234 gives more specific information on the fragility derivation, and particularly the various documents for different reactor designs on which the containment fragility is based for the UK EPR. The two principal references within Ref. EA234 are Refs EA235 and EA236. It should be noted that these documents were developed specifically for the Olkiluoto 3 project, not for FA3 or for the UK EPR. Background information into the phenomenological aspects of the containment challenges is to be found in Ref. EA236.

857 Guidance on the methodology used to determine the fragilities has been taken from Refs OD53 and OD54. These documents are considered to offer current best practice in this area.

858 From reading of Section 3 of Ref. EA234 it is evident, unlike the majority of the UK EPR GDA submission by EDF/AREVA, that Flamanville 3 (FA3) data is not being used. This is because the analysis work is still ongoing and because the FA3 work has not been released by EDF for AREVA to use. Specifically *"For FA3, a structural analysis has been performed, but has not been released by EDF for use in this study"* (Ref. EA234, Section 3.2).

859 Furthermore, it is evident that only limited treatment of leakage failure modes has been considered: "At the present time, not all the planned structural analysis is available. In particular, on-going structural analysis, for which results are not yet available, is investigating containment penetrations and potential leakage failure modes of the containment" (Ref. EA234, Section 3.1).

860 The data as presented for the UK EPR has been derived from the US EPR project, which in turn was derived from data developed for the OL3 project. This convoluted path does not aid clarity in assessing the suitability of the information for inclusion in the PSA level 2 models. TQ-EPR-1161 provided some further clarification of the approach adopted, however there has been no direct link to the FA3 overpressure analysis.

861 A rupture failure mode corresponds to a failure size with an associated rapid depressurisation of containment whereas a leakage failure mode corresponds to a failure size with a slow depressurisation of containment. The US EPR studies assumed '*that the only failure mode is a rupture large enough to depressurise the containment and a leakage fragility curve was neither used nor developed.*'(see Ref. EA234).

862 Fragilities for each failure mode have been defined using a median ultimate pressure capacity and lognormal variabilities for material and modelling variability which are assumed to be independent. The median and 5% probability of failure pressures are presented for each failure mode, and the results from each failure mode are then combined into a composite fragility curve. For a parameter with variability, the approach adopted has been to typically define a median value, and also a value at some other stated NEP. The NEP values chosen were usually the 95% value, but occasionally a higher value has been adopted

863 The following parameters with aleatory or epistemic variability have been identified:

- Material properties (a major source of uncertainty is the expected strain resulting in failure).
- Modelling assumptions.
- Postulated failure criteria.

864 The methodologies for establishing the stress/ strain state of the containment are either by simplified hand calculations for typical zones or by extracting data from FE analyses for more complex singular zones.

865 The six rupture failure modes that have been considered by AREVA for OL3 are as follows:

- Hoop membrane failure in typical zone of the cylinder wall.
- Membrane failure in typical zone of dome.
- Flexural failure of the base of the cylinder wall.
- Flexural failure of dome belt.
- Flexural failure around equipment hatch (vertical section V2).
- Flexural failure around equipment hatch (horizontal section H2).

866 The failure locations for typical sections appear to be reasonable i.e. typical section of the cylindrical wall and the dome. The failure locations for non-typical (singular) sections include the gusset, dome belt and equipment hatch opening. However, there is no discussion regarding which opening is governing. Other potential failure locations could include one associated with buttresses, perhaps at their top level termination point, below

the dome belt, and a failure location at an opening other than the equipment hatch, such as at the personnel hatch which is located within a buttress. In addition, all the failures are for what are essentially ductile type modes. There is no consideration of shear failures, which, whilst they may have greater median values due to the greatly increased uncertainty in mechanical modelling will have different fragility curve slopes, the impact of which cannot be simply assessed on a median basis.

867 The approach of not deriving a bespoke leakage fragility is not necessarily incorrect, however the arguments as presented in Ref. EA234 are not considered sufficiently robust, thus undermining the PSA.

868 The fragility calculations submitted for review relate to OL3. They were used as the basis of the US EPR containment fragility. However, they are not for the UK EPR reference design, namely that for FA3.

869 This has influenced the extent of the review being undertaken during GDA. However, Coyne et Bellier have undertaken the OL3 containment rupture fragility calculations and are also the designers for FA3. Hence it is appropriate in GDA to review the methods in the presented work, even though the application cannot be directly undertaken in the context of FA3.

870 The detailed review of the derivation of the containment fragilities has highlighted the following concerns:

- The OL3 EPR calculations have been scaled to arrive at US EPR values. No evidence of this process has been provided.
- The material variabilities under elevated temperature are stated to have been re-estimated. In general, it appears that the same material variabilites that are used for the modelling assumptions have been used for the additional variability associated with the elevated temperature effect on the materials. There is no discussion regarding why this approach is considered reasonable.
- Thermal stresses appear to have been neglected in loading conditions with no justification.
- The assumed modelling uncertainty for singular regions (e.g. 0.15 for the gusset), arising from the use of linear models for the extraction of demand loads, requires more discussion/justification.
- The documents submitted for the GDA review of the containment fragility do not reflect the GDA reference design.
- All the forces and moments in the singular zones, upon which rupture fragilities have been based, appear to be from linear FE models without justification of non-linear model behaviour.
- The postulated failure criteria do not take account of the multi-axis stress state of the material. Although the tendons and reinforcing bars are stressed in a generally uniaxial fashion, the liner plate is not.
- There is no justification with regard to dismissal of other potential fragilities.
- The general strains in a typical wall section, governed by the 3% strain limit on the post-tensioned tendons, appear to fall considerably outside the 1% general recommendation. A liner strain of around 2.6% is deduced from the calculation, and raises particular concern for liner integrity. Liner strains are even higher for other

rupture fragility calculation. For example, at the base of the cylindrical wall, the liner strain is deduced to be 11%.

871 The comments above are very similar to those generated for the beyond design basis study for the containment and for the reliability calculations undertaken as part of the response to RO-UKEPR-037. This has been raised as GDA Issue **GI-UKEPR-CE-05**.

872 The containment fragility values used for the Level 2 PSA are considered to be sufficiently accurate at this stage, however a more refined set of fragilities will be required at a site specific stage (Assessment Finding **AF-UKEPR-CE-60**). This is linked to GDA Issue **GI-UKEPR-CE-05.**

873 The following assessment findings have arisen

AF-UKEPR-CE-58: *The licensee shall confirm that relay chatter is not a concern for the proposed plant and equipment for a particular site either through elimination of components which exhibit this behaviour or by suitable testing. This should be completed ahead of delivery to Site of Mechanical. Electrical and C&I Safety Systems.*

AF-UKEPR-CE-59: *The licensee shall confirm that the seismic fragilities used are valid for the particular site conditions. This should be completed ahead of Fuel Load.*

AF-UKEPR-CE-60: *The licensee shall develop a more refined set of containment fragilities for site specific application to the PSA. This is linked to GDA Issue GI-UKEPR-CE-03. This should be completed ahead of Fuel Load.*

4.3.11 Qualification of Equipment against External Hazards

4.3.11.1 Scope

874 There are a large number of plant items for which qualification against external hazards is required. This will be done through a mixture of analysis, testing, or experience data. The key Steps in the assessment process are as follows:

- Establish procedures used for qualification of plant and equipment against external hazards.
- Review against modern standards and DBE expectations focussed on:
  - Use of generic testing data.
  - Applicability of testing regime to anticipated demand.
  - Use of special arguments to justify plant beyond test regime.
  - Application of experience data.
  - Applicability of analysis to plant.
- Establish plant items for which no demonstration of adequacy has been undertaken as yet.

4.3.11.2 Standards

875 The key SAPs which are applicable to this area are as follows.

Engineering principles: equipment qualification	Qualification procedures	EQU.1
Qualification procedures should be in place to confirm that structures, systems and components that are important to safety will perform their required safety function(s) throughout their operational lives.		

Engineering principles: maintenance, inspection and testing	Type-testing	EMT.3
Structures, systems and components important to safety should be type tested before they are installed to conditions equal to, at least, the most severe expected in all modes of normal operational service.		

188 *For components of particular concern and where it is not possible to confirm the ability to operate under the most onerous design conditions, reference data from commissioning or rig testing should be established for comparison against in-service test results.*

4.3.11.3 Assessment

876 Qualification of equipment is discussed in Sub-chapters 3.1 and 3.6 of the PCSR. Sub-chapter 3.1 Section 1.2.5.6 states the following general safety principles regarding equipment qualification:

- "*The objective of qualification is to confirm that equipment is capable of fulfilling its functions under the postulated conditions to which it may be subjected."*
- *"The qualification approach is required only for safety classified equipment."*
- *"...multiple standards have been defined for the EPR. These multiple standards, termed "families", are used to demonstrate qualification in design accident conditions."*
- "Different internationally recognised methods may be used for qualification, based on RCC-E, KTA or IEEE standards."

877 Sub-chapter 3.1, Section B.2.2.1 states that:

- *"The equipment needed for the demonstration of safety must be qualified for the conditions for which they are necessary".*
- *"The designer must specify his general qualification approach for classified equipment; this approach must be applied to all types of equipment (mechanical, electrical, etc.) in and outside of the reactor building and take account of internal and external accident conditions and ageing."*

878 Sub-chapter 3.6, Section 1.2.2.2.2.3 states that:

- *"A given type of equipment may be used in several different locations throughout the plant and/or may be required to operate in different types of accidents. Such equipment is qualified for the most severe conditions in which it is required. In practice, to qualify a given item of equipment, a profile bounding the profile required …is generally used."*

879 TQ-EPR-837 was raised for EDF and AREVA to provide further information on the proposed approach for equipment qualification. The overarching approach is to leave the qualification to the suppliers of the equipment but to provide them with sufficient guidance and information in terms of demands, and standards to be used, augmented by requirements in terms of documents to be delivered and necessary checks and balances through the process.

880 The documents for the UK EPR have not been developed as yet, however a sample of FA3 documents have been provided to illustrate the process. Ref. TSC44 provides a more detailed assessment of Equipment qualification.

881 TQ-EPR-211 was raised, requesting information on the process of defining the functionality requirements under seismic loading, and where the requirements are described in the PCSR or elsewhere. The TQ response noted that for each PCC and RCC transient considered a functional requirements analysis is performed. This analysis is used to define the system features that are required to mitigate the transient, and to identify the phase of the transient in which the system feature is required. This analysis is used to define the minimum functional classification required from the system features. By examining the safety functions performed in all transients, the most demanding functional classification is identified and assigned to the equipment item

882 The assignment of a particular safety classification category to an item of equipment is dependent on the level of demand for which its safety function is required. Ref. EA149 presents an EPR equipment classification list for FA3 that shows which level of classification for each of the safety classification categories is assigned to each item of equipment.

883 Ref. EA150 gives a list of equipment to be qualified for accident ambient conditions for FA3. For each item of equipment, the functions that are required to be ensured in accident ambient conditions are presented with the corresponding family of ambient conditions that is applicable to each function during that accident situation. The severe accident qualification requirement for each function is also indicated.

884 Ref. EA151 gives the functional requirements of each plant system on FA3. The functional classification is indicated for each item of equipment and the role of each function, the functional requirements and information required to perform the function are discussed. This demonstrates that where safety systems perform more than one function then a separate set of performance requirements is defined for each function.

885 In summary, for FA3 performance specification for normal and fault conditions appears to be defined. In addition, where safety systems perform more than one function then a separate set of performance requirements is defined for each function. As far as can be ascertained at this stage, it is considered that the documents sampled for Flamanville would, if replicated for the UK EPR provide a sound basis on which to start equipment qualification.

886 A sample of the assorted books of technical specifications and the system design manuals has been undertaken, where available.

887 Contracts related to equipment qualification require a technical specification to be generated in-line with the EDF/AREVA procedure '*INS EPR 328*'. The technical specification for each equipment item uses a number of design input sources and is the first phase of the EQ process. These sources are general specifications such as BTRs (Ref. EA148), standards, regulations (Refs EA152 to155) etc, elementary studies such as SDMs, functional characteristics and operating parameters etc, and qualification

requirements such as functional, seismic/safety classification and accidental condition lists etc.

888 On the basis of the technical specification, the supplier is required to establish the EQ which is the second phase of the EQ process. The supplier carries out the assignment of bounding seismic level, in addition to pressure and temperature profile and dose. Different qualification practices are used for EPR, namely RCC-E, KTA and IEEE. According to EDF and AREVA these were all independently shown to be consistent with the international standard IEC 60780 in the 1*990's for the Commission of European Communities*' although no reference is quoted for this.

889 RCC-E has been used by EDF and AREVA for testing of electrical equipment. It is stated in Ref. EA148, that IEC 60980 was '*designed initially for electrotechnical equipment, it also applies to devices and other types of equipment*'. It also states with regard to external hazards that '*the chosen practice is one of those presented for seismic testing of electrical equipment by IEC 60980*'.

890 The requirements of RCC-E and IEC 60980 have been examined, the KTA or IEEE have not been reviewed. The approaches outlined have been found to be broadly acceptable. Qualification by testing, analysis and operating experience are all options which are potentially valid.

891 Spatial interactions are considered by the SC2 class which are assigned when equipment failure could have an unacceptable outcome on SC1 class equipment. Thus, there is evidence here of spatial interactions being considered during the EQ process.

892 Anchorage of equipment is usually carefully specified in order to ensure that the shake table testing is comparing like with like. However, no information has been sighted with regard to anchorage design and installation. Within the derivation of seismic fragilities in support of the seismic margins assessment, the assumption was made that the anchorage would not be a controlling feature. Clearly this is an important aspect to be addressed beyond GDA.

893 From the limited information available at GDA stage, it is considered that the broad approach suggested is acceptable. However, it is not possible to give a definitive view on this at GDA, and it should be addressed at the site-specific stage. The licensee shall develop a set of arrangements for the qualification of plant and equipment against the demands from Internal and external hazards (Assessment Finding **AF-UKEPR-CE-61**).

4.3.11.4 Summary

894 The general principles for equipment qualification outlined in the PCSR appear to be reasonable.

895 The approach suggested is broadly acceptable, however the information thus far provided is Flamanville specific. This will need to be reviewed in detail at a site specific level. The following assessment findings have arisen.

> ***AF-UKEPR-CE-61:*** *The licensee shall develop a set of arrangements for the qualification of plant and equipment against the demands from Internal and external hazards. This should be completed ahead of the installation of the polar crane. Modifications to the mechanical and electrical equipment associated with the polar crane may be necessary.*

4.3.12 Maintenance Inspection and Test

896 The operational lifetime of the EPR is stated as 60 years, with a construction period of 5 years and a likely decommissioning period of 20 years. In order to ensure ongoing functionality of the structures, a programme of maintenance, inspections and tests will be required. It is considered that the bulk of these inspections will be defined as part of the site specific activities.

897 This aspect is not considered any further in this report, with the exception of the decennial test of the containment, which has been addressed in Section 4.3.6.4.4 although of equal importance are the integrated leak rate tests and the penetrations undertaken on a rolling basis at refuelling outages.

4.3.13 Design Process

898 The design process has been examined in parallel with the deep slice evaluations of the design itself. Ref. TSC40 provides further details.

899 The design of the EPR structures has been developed over many years by a series of design teams working in EDF and AREVA. Engineering support for the UK EPR GDA Project from EDF – France is provided by the EDF Nuclear Engineering Division (DIN). The Nuclear Engineering Division, which is part of EDF Production Engineering Division (DPI), covers the activities of plant design, construction and commissioning of installations.

The Nuclear Engineering Division comprises the following six organisations:

- CNEN for nuclear design engineering
- CNEPE for electricity production engineering
- CIPN for nuclear base engineering
- CIDEN for decommissioning and environmental engineering
- SEPTEN for thermal and nuclear engineering and projects.
- CEIDRE for inspection and testing appraisal

The CNEN, CNEPE, CIDEN and SEPTEN engineering centres are involved in the pre-licensing phase of the UK EPR project. CNEN have placed contracts directly with Sofinel, to provide the following services in relation to Civil Engineering:

i) The drafting of technical specifications, 2nd phase schedules.

ii) Radiation protection, fire, etc.

iii) Drafting of formwork guide drawings (P10, P13), load drawings.

iv) Production of procedures, hypothesis notes".

900 Sofinel are a management organisation jointly owned by EDF and AREVA, contracted to provide engineering services to EDF and AREVA. For the purposes of this examination, Sofinel are considered a contractor to EDF and AREVA.

901 Sofinel have subsequently placed contacts with IOSIS, Coyne et Bellier, NDA, NFM, and Stangenberg for various aspects of the design of civil structures.

902 In order to gain confidence in the design we have investigated the nature of the contractual arrangements put in place by EDF and AREVA to ensure adequate quality in the design.

903 EDF/CNEN is referred to as a Level 1 engineering organisation.

The EDF/CNEN Management System is based on the following standards:

- Quality Order of 10 August 1984 (consistent with IAEA 50-C-QA)
- ISO 9001 (Quality)
- ISO 14001 (Environment)
- OHSAS 18001 (H&S)

The following QA certification has been observed:

- EDF/CNEN, Montrouge, France, ISO 9001:2000, Certificate No. 8431-2007-FRA DNV
- EDF/CNEN, Montrouge, France, ISO 14001:2004, Certificate No. 2009/33945 AFNOR
- Flamanville, ISO 14001:2004, Certificate No. 2009/33945 AFNOR
- EDF/CNEN, Montrouge, France, OHSAS 18001:1999, Certificate No. 8431-2007-FRA DNV
- Flamanville, OHSAS 18001:1999, Certificate No. 8431-2007-FRA DNV

904 EDF's contractual specifications as regards quality assurance are aimed at obtaining from supplier products that comply with their requirements and offer lasting, reliable use. For this purpose, EDF relies primarily on the ISO 9001 version 2000 standard. It represents the reference for its requirements in terms of quality assurance and serves as a basis for the General Quality Assurance Specifications (GQAS).

Two versions of the GQAS exist. These are:

- The “IFS domain” GQAS which solely concerns products that are “Important for Safety” i.e. involved with nuclear safety.
- The “conventional domain” GQAS which concerns all products not involved in the IFS domain.

905 The IFS GQAS supplements the ISO 9001 in order to comply with the requirements of the decree of 10th August 1984 regarding the quality of the design, construction and operation of basic nuclear installations, which is consistent with the IAEA 50-C-QA. The conventional GQAS is used so as not to impose these complementary requirements on all product contracts. EDF specifies the products to which the “IFS” and “conventional” GQAS apply.

906 SOFINEL is referred to as a Level 2 engineering organisation. The SOFINEL Management System is based on the following standards:

- ISO 9001:2008
- IAEA GS-R-3

Copies of the following certification have been seen:

- SOFINEL, Montrouge, France, ISO 9001:2008, Certificate No. 1999/11689b

- SOFINEL, Erlangen, Germany, ISO 9001:2008, Certificate No. 1999/11689b

907 IOSIS Industries are considered to be a Level 3 engineering organisation. Their Management System is based on the ISO 9001:2008. Certificate No. 2008-32330a was observed.

908 Coyne et Bellier (part of Tractobel) are considered to be a Level 3 engineering organisation. Their Management System is based on the ISO 9001:2008. Their certification was observed. (Certificate No. FR07/0234QU SGS).

909 NFM technologies are considered to be a Level 3 engineering organisation. Their Management System is based on the ISO 9001:2008. Their certification was observed.

4.3.13.1.1 Design Process – Assessment

910 The CNEN surveillance programme is planned according to a "surveillance programme" based on contractual requirements (Ref. EA248). Surveillance of subcontracted activities is based on elements contractually available to CNEN, such as *"the list and forecast distribution list of design documents and the documents themselves (study reports, drawings, specifications)".*For each entity in the Preliminary Documentation List (LPD), the surveillance programme defines both the scope and level of the surveillance. With regard to the scope of the surveillance, the term "Modulation" denotes that not all of the supplied documents are examined.

911 The scale and scope of the surveillance carried out in particular take account of:

- *"Quality requirements defined for the project,*
- *The importance of the consequences of any error in design study, in particular for safety or availability,*
- *The level of complexity and expertise in the deign studies,*
- *The need to ensure a level of consistency for all the design studies and the construction,*
- *The irreversibility of the design study or the possibility of detecting error at a later date without generating cost overruns of delays for the project."*

The surveillance programme defines the level of surveillance undertaken. The three levels of surveillance referred to are identified below..

In general, the levels of verification (for reports) involve the following checks:

<u>Level 1</u>

- Suppliers comply with their own QA requirements.
- Suppliers comply with EDF format requirements.
- Suppliers address remarks made by EDF (or SOFINEL).
- The authoring party follows revision guidelines.

Level 2

Level 1 + verification of the overall coherence of design hypotheses, method and conclusions.

The hypotheses comply with:

- EDF Positions/Actions or Commitments.
- EPR Project Technical Directives.
- EDF experience feedback.
- The convergence process.
- The PSAR.
- The Level 1 baseline (i.e. the Site Data, General Hypothesis Note and ETC-C).
- The available interface data.
- The verification of material characteristics, loadings and load combinations, and the geometrical and functional characteristics of the structure.
- The verification of the application and conformity to the calculation methodology defined in supporting documents.
- The verification of the qualification of the codes/programs/procedures used to establish the results.
- The verification of the completeness of the results with regard to the objectives defined in the contract.

Level 3 (calculation notes only)

Level 1 + Level 2 + independent verification. Mainly used when the results are contested or incomplete and the Question Response Forms do not provide conclusions on technical divergences.

- Analysis of the results of a part of the note using simplified methods or by transposition of existing comparable designs to check the orders of magnitude.
- A re-calculation if necessary.

912 In order to examine the operation of the process, the treatment of General Hypothesis Note ECEIG021405 (Ref. EA12) and its Revisions was examined in detail. TQ-EPR-536 requested details over the procedures put in place to ensure that changes at a high level are captured in lower level documentation. The response stated that *"Each issue of a high level document such as the General Hypothesis Note is formally transmitted to SOFINEL by a modification sheet as prescribed in instruction INS.EPR.313. If need be the transmission can highlight major changes, which could have consequences on lower level documents and require a new issue of these documents. In all cases, SOFINEL assesses the need for a new issue of the lower level documents".*

913 Examination of the documents on EDF's document control system (part of "Documentum") showed that Issue F had several non-trivial changes and had been signed on the 22 February 2010. CNEN reported that Issue F had been sent to SOFINEL in March 2007, requesting that it be issued to its contractors. As the transmittal sheets could not be located during the meeting TQ-EPR-830 was subsequently raised.

914 The response to TQ-EPR-830 states that:

> *"In the specific case of the status change from revision E to F, the general hypothesis note was sent to SOFINEL solely with an accompanying letter. The new revision was made to clarify some hypothesis after monthly meetings with SOFINEL and design offices. There were no major changes that could have an impact on the studies made on the nuclear island (cf. ECEP070318).*
>
> *The modification sheet as described in TQ-EPR-536 is a specific contract document made to detail the changes in the EPR design sent to SOFINEL. The aim of this document is to determine the impact on the initial contract. The modification details the changes and some of the actions to be taken:*
>
> - *SOFINEL has to work according to the changes.*
> - *SOFINEL has to transmit the changes to the design offices.*
> - *IGC has to update the general hypothesis note, which will be sent later with an accompanying letter.*
>
> *When the changes in a document do not have any contractual impact, the general hypothesis note is sent to SOFINEL with an accompanying letter such as revision F."*

915 The response to TQ-EPR-830 indicates that a modification sheet is not required if the changes in a document do not have a contractual impact. This was not made clear in the response to TQ-EPR-536. TQ-EPR-536 inferred that every issue of a high level document was transmitted to SOFINEL using a modification sheet.

916 It is clear that Issue F had several non-trivial changes. It would therefore appear that it was judged by COB that the latter "non-trivial" changes had no impact on the EPR design. During our inspection, the transmittal letter from SOFINEL to COB (letter SFL/EF/GI/2007.193 15 March 2007) was located in COB's office records. This letter arrived at COB's head office in Paris on 19 March 2007 according to the date stamp. It was also annotated in Paris to indicate that it should be copied to the COB office in Lyon. No further marks were added to the letter upon receipt at the COB office.

917 Issue F of the General Hypothesis Document was examined and found to contain "sidelines" indicating revisions to the text. No comments or markings were found on the document to indicate that implications of the changes had been recognised. The document did however have a "post-it" note stuck to one of the revised pages.

918 It was evident that some action did result from receipt of the Issue F General Hypothesis Note because Issue F was referenced on COB's own hypothesis note 11815 28B03 NT 003 D, issued on the 20 March 2007. Issue D of COB's hypothesis note was however issued in response to a SOFINEL Observation.

919 It is therefore clear that arrangements for the control of revised documents and their implications on the project are not as rigorous as would be expected. This is captured in the assessment findings and issues within the MSQA topic report.

920 The SOFINEL surveillance programmes were investigated via COB document 11815 28 B03 0032 which was chosen at random. This calculation note is associated with the reinforcement around the emergency air lock close to the Fuel Building. CNEN reported that this calculation appears in their surveillance programme. This was confirmed by inspection of document ECEIG061114 B1. The surveillance programme shows that calculation 11815 28 B03 0032 requires verification to Level 2 or 3. CNEN stated that

Level 3 verification is performed depending on the Level 2 findings. CNEN subsequently pointed to the following statement in Ref. EA249.

921 Section 4.2.3 Surveillance of Calculation Notes and Dimensioning Reports:

Depending on the results of (Level 2) *surveillance or the specific Project requirements, these notes may be the subject of level 3 surveillance, in particular when the results are disputed of incomplete and the Question – Answer Sheets do not enable technical discrepancies to be directly resolved. This in-depth surveillance consists in:*

- *A level 1 and level 2 check*
- *Analysis of the results of a part of the note using simplified methods or by transposition of existing comparable designs to check the orders of magnitude*
- *A recalculation if necessary.*

For notes covered by surveillance control, the same level of surveillance is applied, but also comprises an analysis of the exhaustiveness of the SOFINEL surveillance of these notes (simply by comparison of the comments). By default, all the points opened and covered in the Question and Answer sheets transmitted to the IGC department, require particular surveillance and can even be the subject of level 3 surveillance. The surveillance is generally formalised by observation sheets."

922 CNEN were requested to provide evidence of the review performed by CNEN on Calculation Note 11815 28 B03 NT 0032. This evidence was tabled by way of a spreadsheet (20100315_SuiviYR1223.xls). Examination of the spreadsheet showed that only one entry ("Référence remarques IGC" against Issue D) related to CNEN's surveillance of the document. For all document versions it was not clear whether a review had been performed, or a review had been performed without any comments arising. CNEN stated that it was believed that the blank entries were because CNEN had no further comments beyond those raised by SOFINEL.

TQ-EPR-713 was raised to answer the following questions:

- Are there any controlling QA procedures which specify how findings such as "no comments" or "no further comments beyond those raised by SOFINEL" should be recorded in the spreadsheet?
- When a review finds "no comments" or "no further comments beyond those raised by SOFINEL", what records are held to verify that a review was carried out, by whom and when? If records are available, please provide evidence relating to the review of Calculation Note 11815 28 B03 NT 0032 Issue B.

923 With regard to the first point, the response to TQ-EPR-713 states that the spreadsheet is an internal document which enables the project engineers to follow the surveillance of documents issued under contract YR 1223. It grows with the surveillance activity and acts as a basis of the oversight folder. TQ-EPR-831 however states *"These spreadsheets are used by engineers (EDF and SOFINEL) to keep a record of surveillance. Engineers are responsible for their files and they are not subject to QA."*

924 There is a concern that these spreadsheets are not subject to QA. As stated in the response to TQ-EPR-831, the spreadsheets can be *"considered as a record of the surveillance made since the beginning of the contract to the present day."* These spreadsheets are an important part of the surveillance programme, and it not evident what other controls are being claimed as a replacement.

925 With regard to point two, the response states: *"If there are no comments to add to those of SOFINEL, finalisation awaits completion of the review". "In the contract, this review period is limited to 28 days, after which, the Design Offices are allowed to issue a final version. It is still possible to add comments at this stage, but a re-issue would then be necessary."*

926 This infers that if EDF either (a) has "no comments" having reviewed the document or (b) has not reviewed the document within the allotted time frame i.e. 28 days, the document is automatically issued by the Design Office. In the latter case it is clear that no records will be held. In the former case the response is unclear as to whether records are held regarding the review process.

927 With regard to the surveillance of Calculation Note 11815-28-B03-NT-0032/B the response states: *"No specific surveillance document had been issued by EDF with regard to revision B PREL of document 11815-28-B03-NT-0032 during the first 28 days. It had been read by SOFINEL and the SOFINEL observation form has been verified before being sent to COB. The report was accepted."*

928 The response indicates that the document was not reviewed by EDF and hence issued automatically after 28 days. As such, no records would be held regarding the review process.

929 No further concern is raised here, beyond the more general one already stated that the spreadsheet should be part of the QA process. This is based on the 28 day rule effectively being a commercial issue. Comments may still be raised after 28 days, arising from a late review.

930 SOFINEL's surveillance of COB's work by SOFINEL was sampled using COB's Calculation Note 11815 28 B03 0032.

931 SOFINEL tabled spreadsheet "YR1223_programme_surveillance_28.xls". Inspection of the spreadsheet showed an entry of "O" against Issue B of the calculation note. SOFINEL stated that "O" indicates "sans object" (in French) or "no comment" (in English). The date of the entry was not recorded and hence it was not clear whether it had resulted from a review of the document or had been assigned automatically after 28 days.

932 With regard to the records of the review, SOFINEL reported that the QA system requires that a review sheet should be produced when a document is reviewed; even if no comments arise. SOFINEL acknowledged however that a review sheet was not always produced when there were no comments.

933 To address these quality issues TQ-EPR-714 was raised. The TQ requested clarification of the following points:

- Are reviews always performed, even when the 28 day deadline is exceeded?
- How many reviews were not undertaken within 28 days?
- Give details of how the above questions have been answered with regard to the source information.

934 With regard to the first point, the response states that *"all contract YR 1223 document references (1 document reference concerns 2 or more revision indexes) have or will be reviewed".* For all documents with BPE status, SOFINEL has confirmed that the observations were taken into account and a record made in the YR 1223 surveillance follow-up sheet with an entry "O".

935 With regard to the second point, the TQ response states that a large percentage of documents (53% of reports) were not reviewed within 28 days. However, of the large percentage of documents not reviewed within 28 days there were only a few instances of documents being assigned BPE status automatically after the 28 day period. Whilst recognising that a large percentage of reviews were not performed by SOFINEL within the 28 day period, there were no instances where the supplier did not take into account the feedback from SOFINEL.

936 It is clear from the samples undertaken above that subcontractor control has not been undertaken to the level expected for design of Nuclear facilities. It is therefore clear that there is a need to undertake sufficient confirmatory reviews for existing work and secondly and to put in place suitable procedures for control of any new works for site specific activities or for new work undertaken in support of resolution of GDA Issues (Assessment Finding **AF-UKEPR-CE-62**). The need to ensure appropriate subcontractor control for site specific activities is captured in the findings under the MSQA topic area.

937 The internal verification undertaken by COB was reviewed. TQ-EPR-710 was raised to request more information on these levels of verification. In their response COB stated that three *types* of control are defined in procedure DTE-01. They are:

- Type 1 – Self-check. The control is realised by the author. The value of the self-check is strengthened by the decision of the author to consult, when he considers it useful, a more qualified person or more experienced person, to confirm his/her methods of work and the way to establish the document.
- Type 2 – Internal control. The control is performed by a specialist engineer, not the author of the document, but belonging to the project team. The checker is nominated, following the organisational structure of the project team, by the Director of the Unit or Project Manager, in agreement with his superiors.
- Type 3 – Independent control. The control is performed by either an expert of COB, nominated by the Director of the Unit or Project Manager, in agreement with his superiors or an external organisation nominated in the Quality Plan

938 We have established that NFM do not have different levels of internal verification. The details of the verification level were not presented. This is a concern given the safety critical nature of the work.

939 IOSIS reported that there was one level of verification carried out by "experienced personnel". This is a concern given the safety critical nature of the work.

940 The presence of SQEP registers and information on training records of staff was sampled for COB, NFM and IOSIS. The results were mixed, with some organizations having formal arrangements and others relying on CV's for evidence.

941 The following Assessment Findings have emerged:

> ***AF-UKEPR-CE-62:*** *Where a licensee places reliance on work previously undertaken by subcontractors to EDF and AREVA they shall undertake sufficient confirmatory reviews for existing work and secondly and to put in place suitable procedures for control of any new works for site specific activities. This should be completed ahead of the placement of first structural concrete.*
>
> ***AF-UKEPR-CE-63:*** *Not used.*

4.3.13.1.2 Design Process – Findings

942 In summary,

- The arrangements in organisations were not considered sufficiently robust for design of nuclear related structures. The individuals involved and the outcomes were found to be adequate, however this was not a direct result of the systems and processes in place.
- There will be a requirement to implement a much more stringent set of arrangements on subcontractors employed for site specific activities in the UK EPR. This is covered by the assessment findings and GDA Issues covered under the MSQA topic area.
- The licensee shall undertake sufficient confirmatory reviews for existing work and secondly and to put in place suitable procedures for control of any new works for site specific activities or for new work undertaken in support of resolution of GDA Issues.
- Where a licensee places reliance on work previously undertaken by subcontractors to EDF and AREVA they shall undertake sufficient confirmatory reviews for existing work and secondly and to put in place suitable procedures for control of any new works for site specific activities (Assessment Finding **AF-UKEPR-CE-62**). This should be completed ahead of the placement of first structural concrete.
- There is no evidence of any form of independent nuclear safety assessment of any aspect of the design of the civil structures. Within the UK regulatory context this is considered a requirement. As a consequence any licensee shall have in place arrangements to undertake independent nuclear safety evaluation of the design of safety critical structures. The following assessment finding has been raised:

  AF-UKEPR-CE-64: *The licensee shall have in place arrangements to undertake independent nuclear safety evaluation of the design of safety critical structures. This should be completed ahead of the placement of first structural concrete.*

4.4 Overseas Regulatory Interface

943 HSE collaborates with overseas regulators, both bilaterally and multinationally.

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

- the US Nuclear Regulatory Commission (US NRC)
- the French L'Autorité de Sûreté Nucléaire (ASN)
- the Finnish Säteilyturvakeskus (STUK)

945 We have had limited interactions with ASN (Refs ND22 to ND24). We have received copies of some documents which they have used in their assessment. (Ref. OD33). We have held one bilateral meeting towards the end of the work program. The general impressions from that meeting were as follows.

- The concerns we have expressed over subcontractor control and surveillance were shared by ASN.
- The most recent version of ETC-C (AFCEN 2010) was seen, particularly part 2 as being less restrictive than ETC-C Revision B.

946 Currently, Flamanville 3 in France and Olkiluoto 3 in Finland are under construction and as a result have had aspects of the design of their Civil Engineering reviewed by their respective regulators ASN and STUK.

947 In addition, the US NRC in the United States is currently reviewing the US EPR against their requirements.

948 We have undertaken a review of publicly available documents from ASN and STUK. These are discussed further in the following paragraphs.

949 As part of the authorisation to build Flamanville, ASN produced an executive summary of the technical review (ASN/DCN/Report No. 0080-2007). This identified assorted Groupe Permanante reports and meetings that documented the review of the EPR. As a result, the following reports were highlighted for further consideration; EPR Report Nos, 87,88,89,90 and 91. These are also known as DSR Reports 18, 34, 69, 2 and 103.

950 Thus far only DSR report No. 69 (provided under Ref. OD33) has been examined in any detail. It discusses in broad terms the design approach to the EPR civil structures. The key conclusions reached in this report are as follows;

- The ETC-C (Rev B) meets the intent of the ASN technical guidelines, however there are some clarifications over the design load cases around guillotine failure of aspects of the primary circuit.
- The proposal to reduce the test pressure of the containment is not considered acceptable.

951 STUK issued a statement into the public domain in January 2005 entitled 'Safety assessment of the Olkiluoto 3 (OL3) Nuclear power plant Unit for The Issuance of Construction Licence' (Ref. OD13) This document whilst high level in its content has some useful comments to make on the containment integrity and on the aircraft protection shell.

- The containment at OL3 is fitted with a venting arrangement, which is filtered. This is a requirement of Finnish regulations. In the UK, we have no such prescriptive requirement, and the design does not incorporate a vent system for the containment.
- The cooling water system has been provided with an alternative intake route for sea water should the main intakes become clogged by ice, sea borne debris or organic matter.
- The aircraft protection shell has been assessed against the demands from a large passenger jet and the subsequent fuel fire and the safety justification found to be adequate.

952 The bulk of our collaborative effort with ASN has been through a review of their official letters to EDF on the Flamanville 3 project. These are focused on the implementation of the design rather than the actual design itself. They have been reviewed however to ascertain if there are any lessons that can be learnt for the design itself. The following paragraphs give a brief synopsis of the findings.

953 The French regulator ASN undertook a site inspection at Flamanville in December 2007. This highlighted a number of concerns including:

- Control of concrete heat of hydration tests.
- Control of water cement ratios.
- Control of the concrete cube samples collected.

- Adequacy of adherence to the construction quality plan.
- Lack of space to fit 'Tremi' tubes into the reinforcing cage and subsequent higher drops than permitted.

These findings were transmitted to EDF by ASN in Letter DEP-CAEN-0045-2008.

954 A second inspection by ASN occurred on 8th February 2008. This highlighted the following issues:

- Qualification of workshop carrying out welding of the liner plates.
- Completeness of the welding log for the liner.
- Control of crack injections on the base slab.
- Unexpected water ingress to the pre-stressing gallery.

These findings were transmitted to EDF by ASN in Letter Ref: Dep-CAEN-0117-2008.

955 A third inspection by ASN occurred on 5 March 2008. This highlighted concerns in the following areas:

- Control of steel reinforcement placement, inspection and treatment of non-conformances.
- Consistency between the work plan and the implementation.
- Positional checking for inserts following concreting.

These findings were transmitted to EDF by ASN in Letter DEP- CAEN-No.0185-2008

956 In addition, cracking in the base slab of the reactor was observed following the first level pour. This has been caused as a result of a lack of anti-crack reinforcement at the top of the lower level pour.

957 In 2009, there have been a further 7 inspections at Flamanville, and there have been a series of observations of non-conformity with the requirements of part 2 of ETC-C, which specifies the construction requirements. In each case, EDF have been requested to amend their procedures to ensure compliance with the intended procedures.

958 A number of issues relating the quality of construction at Olkiluoto 3 were revealed during the early construction phases. STUK (The Finnish Nuclear Regulator) undertook an inspection of these and reported them in Ref. OD14. The key conclusions were as follows:

- The concrete as designed was unsuitable for pumping, and modifications were made mid way through pours to allow them to continue. The initial trials were seen as insufficient, and control over modifications to concrete mixes unacceptable.
- The contractor was often poorly prepared for large pours with insufficient workforce.
- The control of the welding of the containment liner was poor.
- The arrangements for the design and fabrication of the hatch and polar crane when audited were found to be using outdated information and standards. The general transfer of information between the fabricator and the designer was seen to be below the standards expected.

959 The bulk of the comments above are linked to the implementation of the design rather than the actual design itself. For the purposes of GDA, there is nothing in the ASN inspections which gives rise to any concerns over the design of the EPR.

960 We have engaged with STUK on the use of the ETC-C for the OL3 design. STUK did not consider that the ETC-C was in line with Finnish building code. As a result, they requested an EPR specific National Annexe type document be prepared (at that time the Finnish NA was not approved). The approaches in this document were typically more stringent than the now finalised Finnish NA. A comparison of the Finnish NA with the UK NA has shown they are broadly similar, with the Finnish NA potentially more conservative in some areas. It can be concluded therefore that provided the ETC-C AFCEN is broadly consistent with the UK NA and Eurocodes then the design rules for OL3 will be broadly similar to those in the UK.

961 Bilateral discussions have been held with the USNRC on the topic of grouted in place tendons. Grouted in place tendons are a novel feature in the US nuclear industry, with all commercial plants with pre-stressed containments having unbonded tendons. The option to use grouted tendons has always been available in the US, indeed two NUREG guides (OD34 and OD35) have been available for some time. These guides are at a relatively high level, and they are somewhat outdated. As a result, the NRC have set about revising them. This process is currently underway. The discussions with NRC were focussed on understanding the concerns HSE had with the EPR design and the resolutions paths we had followed.

4.5 Interface with Other Regulators

962 During the regulation of site specific facilities, there is a need for regular and structured interfacing with the Environment Agency on matters around flood risk and discharges. At the generic stage, flooding from external sources (sea, rain) cannot really be considered in a meaningful way.

4.6 Other Health and Safety Legislation

4.6.1 The Construction (Design and Management) Regulations 2007

963 During Step 2, it was noted that there was no reference within the documentation available to the Construction (Design and Management) Regulations 2007 (CDM 2007). The CDM regulations are somewhat unique in that they apply to projects well ahead of implementation, indeed neither a client nor a commitment to build is required.

964 Regulation 11 states that the duties of designers are:

> ***11.***—*(1) No designer shall commence work in relation to a project unless any client for the project is aware of his duties under these Regulations.*
>
> *(2) The duties in paragraphs (3) and (4) shall be performed so far as is reasonably practicable, taking due account of other relevant design considerations.*
>
> *(3) Every designer shall in preparing or modifying a design which may be used in construction work in Great Britain avoid foreseeable risks to the health and safety of any person;*
>
> *(a) carrying out construction work;*
>
> *(b) liable to be affected by such construction work;*
>
> *(c) cleaning any window or any transparent or translucent wall, ceiling or roof in or on a structure;*
>
> *(d) maintaining the permanent fixtures and fittings of a structure; or*

(e) using a structure designed as a workplace.

(4) In discharging the duty in paragraph (3), the designer shall;

(a) eliminate hazards which may give rise to risks; and

(b) reduce risks from any remaining hazards,

and in so doing shall give collective measures priority over individual measures.

(5) In designing any structure for use as a workplace the designer shall take account of the provisions of the Workplace (Health, Safety and Welfare) Regulations 1992 which relate to the design of, and materials used in, the structure.

(6) The designer shall take all reasonable Steps to provide with his design sufficient information about aspects of the design of the structure or its construction or maintenance as will adequately assist/

(a) clients;

(b) other designers; and

(c) contractors,

to comply with their duties under these Regulations.

In the case of EDF and AREVA, Regulation 12 also applies as below.

965 Regulation 12 states that:

12. *Where a design is prepared or modified outside Great Britain for use in construction work to which these Regulations apply;*

(a) the person who commissions it, if he is established within Great Britain; or

(b) if that person is not so established, any client for the project,

shall ensure that regulation 11 is complied with.

966 It might not be reasonable to expect a design which originated in France to be fully cogniscent of UK regulations at this stage, however it is worth noting that the origin of the CDM regulations is a European Directive. Council Directive 92/57/EEC of 24 June 1992 on the implementation of minimum safety and health requirements at temporary or mobile construction sites provides a pan European directive, which CDM regulations enact in UK legislation. France has developed their own national execution measures through a series of 'Arrêté ministériel'. It would therefore be reasonable to assume that considerations similar to those in the CDM regulations have been given by the designers of the EPR for Flamanville. In addition, Finland has also enacted the directive through a series of Legal acts and regulations and again, it would be reasonable to expect that the designers of the Olkiluoto EPR would have undertaken similar considerations.

967 EDF and AREVA Letter EPR00059 provided some further details on the approach adopted for the EPR. The legal obligation is based in French decree 93-1418, which is the implementation of Council Directive 92/57/EEC. This is also supplemented by Decree 94-1159. (Ref. OD36).

968 Letter EPR0059 goes on to note that a safety coordinator was appointed as part of the design team, and that a dossier of information was derived as part of the layout studies. It is also claimed that reviews at key steps in the design process have been undertaken to ensure that the building is capable of being constructed and maintained safely.

969 I have not conducted a more detailed investigation of these claims during GDA. It is an assessment finding that suitable arrangements are put in place within a site licensee organisation to review the designs from a CDM perspective:

> ***AF-UKEPR-CE-65:*** *The licensee shall ensure that due consideration is given to the CDM regulations for the design of all civil structures (safety and non-safety). This should be completed ahead of the placement of first structural concrete in line with the requirements of UK law.*

5 CONCLUSIONS

970 This report presents the findings of the Step 4 Civil Engineering and External Hazards assessment of the EDF and AREVA UK EPR reactor.

971 To conclude, I am broadly satisfied with the claims, arguments and evidence laid down within the PCSR (Ref. EA252) and supporting documentation for the Civil Engineering and External Hazards as listed in the Submission Master List (Ref. EA253). I consider that from a Civil Engineering and External Hazards view point, the EDF and AREVA UK EPR design is suitable for construction in the UK. However, this conclusion is subject to satisfactory progression and resolution of GDA Issues to be addressed during the forward programme for this reactor and assessment of additional information that becomes available as the GDA Design Reference is supplemented with additional details on a site-by-site basis.

5.1 Key Findings from the Step 4 Assessment

972 The assessment of the UK EPR has focussed on three main areas, design codes, computer codes and the design process.

973 The classification of the safety critical structures has been found to be satisfactory, and the associated design processes suitable. As part of a cross cutting GDA issue (**CC-UKEPR-001)**, there is a need to clarify the classification of lower tier structures.

974 The identification of hazards, and subsequent development of the load schedules for the civil structures has been examined as far as practicable without a specific site being identified, and found to be satisfactory.

975 The main design code used, the ETC-C has undergone significant change during the course of the assessment process. Further justification for ETC AFCEN 2010 and its UK companion document is required: as a result, a GDA Issue has been raised.

976 The computer codes used in the design have largely been found to be satisfactory, with a small number of remnant concerns which may need to be addressed by a future licensee. The analysis approach has generally been found to be adequate, however there are some remaining areas of concern with the methodology for seismic analysis, and a GDA issue has been raised.

977 The design process has been examined at a number of different levels. The design hypothesis notes which were developed for the Flamanville 3 project have been found to need revision for use in the UK, and a GDA Issue has been raised over the development of UK specific documents.

978 There have been a number of deep slice evaluations of the design of elements of individual structures. These have been generally found to be adequate, however in some cases, there are findings which will need to be resolved by future licensees.

979 The design of the inner containment structure has been found to be well thought out and generally acceptable. There are some residual aspects on the analysis of the containment and the beyond design basis performance which require further clarification and have been raised as GDA Issues.

980 There are two key areas where the design process has been challenged: to demonstrate the reliability of the design by ETC-C and to demonstrate the beyond design margin for key safety-related components. Both of these areas have not been completed satisfactorily during the Step 4 of GDA and as a result GDA Issues have been raised.

981 The design against accidental and malicious aircraft impact has been found to be satisfactory.

982 There are a large number of areas where clarification of the design approach in detail will be required by future licensees, however in many areas, the principles have been agreed as satisfactory.

5.1.1 Assessment Findings

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

5.1.2 GDA Issues

984 I conclude that the following GDA Issue(s) listed in Annex 2 must be satisfactorily addressed before Consent will be granted for the commencement of Nuclear Island safety-related construction.

- Hypothesis notes for the Nuclear Island (NI) Structures GI-UKEPR-CE-01.
- ETC-C AFCEN and the UK companion Document GI-UKEPR-CE-02 and 05.
- Beyond Design Basis behaviour of the Inner Containment GI-UKEPR-CE-03
- Analysis of the Inner Containment GI-UKEPR-CE-04
- Reliability of the ETC-C AFCEN GI-UKEPR-CE-05
- Methodologies for the treatment of SSI and seismic analysis GI-UKEPR-CE-06.

985 The complete GDA Issue identified in this report and their associated actions are formally defined in Annex 2.

6 REFERENCES

6.1 Internal References

ND1 *Generic Design Assessment – Guidance to Requesting Parties Version 2*. HSE. 16 July 2007 www.hse.gov.uk/nuclear/meetings/spentfuel09/report.pdf.

ND2 *ND BMS. Assessment Process*. AST/001 Issue 2. HSE. February 2003 www.hse.gov.uk/foi/internalops/nsd/assessment/ast001.htm.

ND3 *ND BMS. Guidance: Assessment Process G/AST/001– Issue 2 HSE*. February 2003 www.hse.gov.uk/foi/internalops/nsd/assessment/ast001.htm.

ND4 *Step 3 Assessment of the EDF and UK EPR PSA*. Accounting for EDF and AREVA comments. AR 09/027. November 2009. TRIM Ref. 2009/409629

ND5 *Safety Assessment Principles for Nuclear Facilities*. HSE 2006 Edition. Available on the HSE website

ND6 *ND BMS. Structural Integrity Civil Engineering Aspects*. T/AST/017 Issue 2. HSE. March 2005 www.hse.gov.uk/foi/internalops/nsd/tech_asst_guides/tast017.pdf.

ND7 *ND BMS. External Hazards. T/AST/013 Issue 3*. HSE. April 2009 www.hse.gov.uk/foi/internalops/nsd/tech_asst_guides/tast013.htm.

ND8 *Step 3 and 4 Civil and External Hazards Assess*ment. Assessment Report No 8027 HSE-ND TRIM Ref. 2008/253146.

ND9 *Step 2 Siting, External hazards and Civil Engineering Assessment.* Assessment Report No 8002. HSE-ND. TRIM Ref. 2008/40647.

ND10 *Unit 6E Operating Plan of Work 2008/09*. HSE-ND. TRIM Ref. 2008/215106

ND11 *GDA Step 4 Civil Engineering and External Hazards Assessment Plan for the UK EPR.* AR 09/061. TRIM Ref. 2009/466637.

ND12 *ND BMS. External Hazards. T/AST/013 Issue 3*. HSE. April 2009 www.hse.gov.uk/foi/internalops/nsd/tech_asst_guides/tast013.htm.

ND13 *Progress on ETC-C. Letter from ND to UK EPR Project Front Office.* EPR70103R. 31 July 2009. TRIM Ref. 2009/297255.

ND14 *Meeting to discuss FE Codes used in the EPR Design*. EDF. Gennvilliers, Paris. 23 July 2009. TRIM Ref. 2009/295421.

ND15 *Meeting with EDF to discuss grouted in place tendons used in the EPR Design*. EDF CNEN. 2 April 2009. Montrouge, Paris. TRIM Ref. 2009/188341.

ND16 *Meeting with EDF to discuss ETC-C Code used in the EPR Design.* EDF SEPTEN. Lyon. 21-22 April 2009. TRIM Ref. 2009/186420.

ND17 *The Response to RO-UKEPR-017 and the way forward*. EDF. Lyon 19 August 2009. TRIM Ref. 2009/347437.

ND18 *EPR Site Preps (Level 4). ND Division 6 Contact Report CR09005.* 20 August 2009. TRIM Ref. 2009/332576.

ND19 *ETC-C User Guide.* Letter from ND to UK EPR Project Front Office. Unique Number EPR70237N. 2 September 2010. TRIM Ref. 2010/384861.

ND20 *Various Submissions on ETC-C.* Letter from ND to UK EPR Project Front Office. Unique Number EPR70291R. 3 February 2011. TRIM Ref. 2011/70303.

ND21 Response to RO-37 Step 2. Letter from ND to UK EPR Project Front Office. Unique Number EPR70288N. 21 January 2011. TRIM Ref. 2011/47921.

ND22 *Co-operation and technical assistance between HSE (NII) and ASN on EPR Nuclear Reactor Assessment*. Letter from ND to ASN. 2 March 2009. TRIM Ref. 2009/85643.

ND23 *Co-operation and technical assistance between HSE (NII) and ASN on EPR Nuclear Reactor Assessment - Topic Area Civil Engineering*. Letter from ND to ASN. 11 November 2009. TRIM Ref. 2009/447697.

ND24 *Co-operation and technical assistance between HSE (NII) and ASN on EPR Nuclear Reactor Assessment - Topic Area Civil Engineering*. Letter from ND to ASN. 20 November 2009. TRIM Ref. 2009/464951.

ND25 *ETC-C AFCEN Part 1*. Letter from ND to UK EPR Project Front Office. Unique Number EPR70304N. 20 April 2011. TRIM Ref. 2011/235790.

ND26 *ETC-C AFCEN Part 2*. Letter from ND to UK EPR Project Front Office. Unique Number EPR70367R. 07 October 2011. TRIM Ref. 2011/514252.

ND27 *Step 4 Internal Hazards Assessment of the EDF and AREVA UK EPR™ Reactor*. ONR Assessment Report ONR-GDA-AR-11-017, Revision 0. TRIM Ref. 2010/581514.

6.2 EDF and AREVA Supplied Documents

EA1 *UK EPR Pre-construction Safety Report – November 2009 Submission.* Submitted under cover of letter UN REG EPR00226N. 30 November 2009. TRIM Ref. 2009/481363 and as detailed in UK EPR Master Submission List. November 2009. TRIM Ref. 2011/46364.

EA2 *Presentation of Approach for Incorporating Hazards in the EPR Project*. ENSN040070-A1. EDF. TRIM Ref. 2011/85569.

EA3 *ETC-C (EPR Technical Code for Civil Works) Part 1.* ENGSGC050076 Revision B. EDF. April 2006. TRIM Ref. 2010/404165.

EA4 *Verification and validation. Response to Civil Structures Topic Meeting action 3.1.* EDF. 4 June 2009. TRIM Ref. 2009/220694.

EA5 *Meeting to discuss grouted in place tendons used in the EPR Design*. EDF. 2 April 2009. TRIM Ref. 2009/188341.

EA6 *Meeting to discuss FE Codes used in the EPR Design*. EDF. 11 June 2009. TRIM Ref. 2009/245316.

EA7 *Nuclear Island - Overall 3D Model General Description - Nodes, Elements and Thicknesses*. EPR FA3 YR1221 28B01-0102. EDF. 30 October 2008. TRIM Ref. 2011/85519.

EA8 *Prestressing System for UK EPR inner Containment.* ENGSGC080361 Revision D. EDF. 26 March 2009. TRIM Ref. 2011/513730.

EA9 *Further Response to Civil Structures RO-UKEPR-017 on Grouted Tendons.* Letter from UK EPR Project Front Office to ND. Unique Number EPR00135R. 10 July 2009. TRIM Ref. 2009/276877.

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OD13 *STUK report 'Safety assessment of the Olkiluoto 3 (OL3) Nuclear power plant Unit for The Issuance of Construction Licence '* 2005 http://www.stuk.fi/ydinturvallisuus/ydinvoimalaitokset/ydinvoimalaitosluvat/viides/en_GB/rakentamislupa/_files/12222632510016763/default/olkiluoto3_rakentamisluvan_turvallisuusarvio.pdf.

OD14 *Management Of Safety Requirements In Subcontracting During The Olkiluoto 3 Nuclear Power Plant Construction Phase*, STUK Investigation Report 1/06. http://www.google.co.uk/url?sa=t&rct=j&q=33.%09stuk%20investigation%20report%201

%2F06%20%E2%80%9Cmanagement%20of%20safety%20requirements%20in%20subcontracting%20during%20the%20olkiluoto%203%20nuclear%20power%20plant%20construction%20phase&source=web&cd=1&ved=0CCMQFjAA&url=http%3A%2F%2Fwww.stuk.fi%2Fydinturvallisuus%2Fydinvoimalaitokset%2Folkiluoto3%2Fen_GB%2Folkiluoto3%2F_files%2F12222632510014949%2Fdefault%2Finvestigation_report.pdf&ei=VH2yTrDQKcH38QOC6MThBA&usg=AFQjCNFgahfRjHVQr7yV0zO0K4VW5rJsDQ

OD15 *Basis of structural design.* EN1990 Eurocode 0.

OD16 *Actions on structures.* EN1991 Eurocode 1.

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OD21 *Design of structures for earthquake resistance.* EN1998 Eurocode 8.

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OD36 *Coordination of risk prevention activities on building and civil engineering sites - Organizing site coordination - Training site coordinators*, Act No.93-1418 of 31 Dec. 1993, Decree No.94-1159 of 26 Dec. 1994 and Orders of 7 Mar. 1995.

OD37 *Design of Fastenings in Concrete*, Comite Euro-international du Beton, Thomas Telford, 1997.

OD38 *Recommended practices for seismic qualification of electrical equipment of the safety system for nuclear generating stations,* IEC 60980.. First edition, 1989-06.

OD39 *RCC-E. Design and Construction Rules for Electrical components of nuclear islands*. December 2005 edition.

OD40 *IEC 60780. Nuclear Power Plants. Electrical Equipment of the Safety System. Qualification*. 1998-10.

OD41 *IEC 68-3-3. Environmental Testing. Part 3: Guidance. Seismic Test Methods for Equipments*. First edition, 1991-2.

OD42 *IEEE Std 323-2003. IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations*. January 2004.

OD43 *Grout for prestressing tendons. Test methods*, BS EN 445:2007.

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OD54 *US Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. Regulatory Guide 1.216. Containment structural integrity evaluation for internal pressure loadings above design-basis pressure*. August 2010.

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OD56 *European Technical Approvals Guideline (ETAG) of Post-Tensioning Kits for Prestressing of Structures* (ETAG 013).

Table 10: Summary of SAPs Assessed in GDA for Civil Engineering and External Hazards

SAP Number	SAP Title	Assessed Category
ECS - Safety Classification and Standards		
ECS.1	Safety Categorisation	S2
ECS.2	Safety Classification of Structures, Systems and Components	S2
ECS.3	Standards	S2
EQU - Equipment Qualification		
EQU.1	Qualification Procedures	S3
EDR - Design for Reliability		
EDR.1	Failure to Safety	S2
EDR.2	Redundancy, Diversity and Segregation	S2
EDR.3	Common Cause Failure	S2
EDR.4	Single Failure Criterion	S2
EMT - Maintenance, Inspection and Testing		
EMT.1	Identification of Requirements	S3
EMT.3	Type-Testing	S3
EMT.6	Reliability Claims	S3
EAD - Ageing and Degradation		
EAD.1	Safe Working Life	S3
EAD.2	Lifetime Margins	S3
EAD.3	Periodic Measurement of Material Properties	S3
ELO - Layout		
ELO.1	Access	S3
ELO.4	Minimise Effects of Incidents	S3
EHA - External and Internal Hazards		
EHA.1	Identification	S2
EHA.2	Data Sources	S3
EHA.3	Design Basis Events	S2
EHA.4	Frequency of Exceedance	S2
EHA.5	Operating Conditions	S2

SAP Number	SAP Title	Assessed Category
EHA.6	Analysis	S2
EHA.7	Cliff Edge Effects	S3
EHA.8	Aircraft Impact	S3
EHA.9	Earthquakes	S3
EHA.11	Extreme Weather	S3
EHA.12	Flooding	S3
EHA.13	Storage of Hazardous Materials	S3
EHA.14	Sources of Harm	S3
EHA.15	Flooding	S3
ECE - Civil Engineering		
ECE.1	Functional Performance	S2
ECE.2	Independent Arguments	S3
ECE.6	Loadings	S2
ECE.7	Foundations	S3
ECE.8	Inspectability	S3
ECE.12	Structural Analysis	S2
ECE.13	Use of Data	S3
ECE.14	Sensitivity Studies	S3
ECE.15	Validation of Methods	S3
ECE.20	In-Service Inspection and Testing	S3
ECE.21	Proof Pressure Test	S3
ESS - Safety Systems		
ESS.18	Failure Independence	S2
ENM - Control of Nuclear Matter		
ENM.3	Accumulation	S3
ENM.7	Retrieval and Inspection	S3
ECV - Containment and Ventilation		
ECV.2	Release Minimisation	S3
ECV.3	Confinement Design	S2
ECV.7	Leakage Monitoring	S3
RP - Radiation Protection		

SAP Number	SAP Title	Assessed Category
RP.6	Shielding	S3
RW - Storage of Radioactive Waste		
RW.5	Storage and Passive Safety	S3
DC - Decommissioning		
DC.1	Design and Operation	S3

Figure 2: Hierarchy of Documents

Level 1 EDF

Level 2 EDF

Level 3 Sofinel

Level 4 (Design offices)

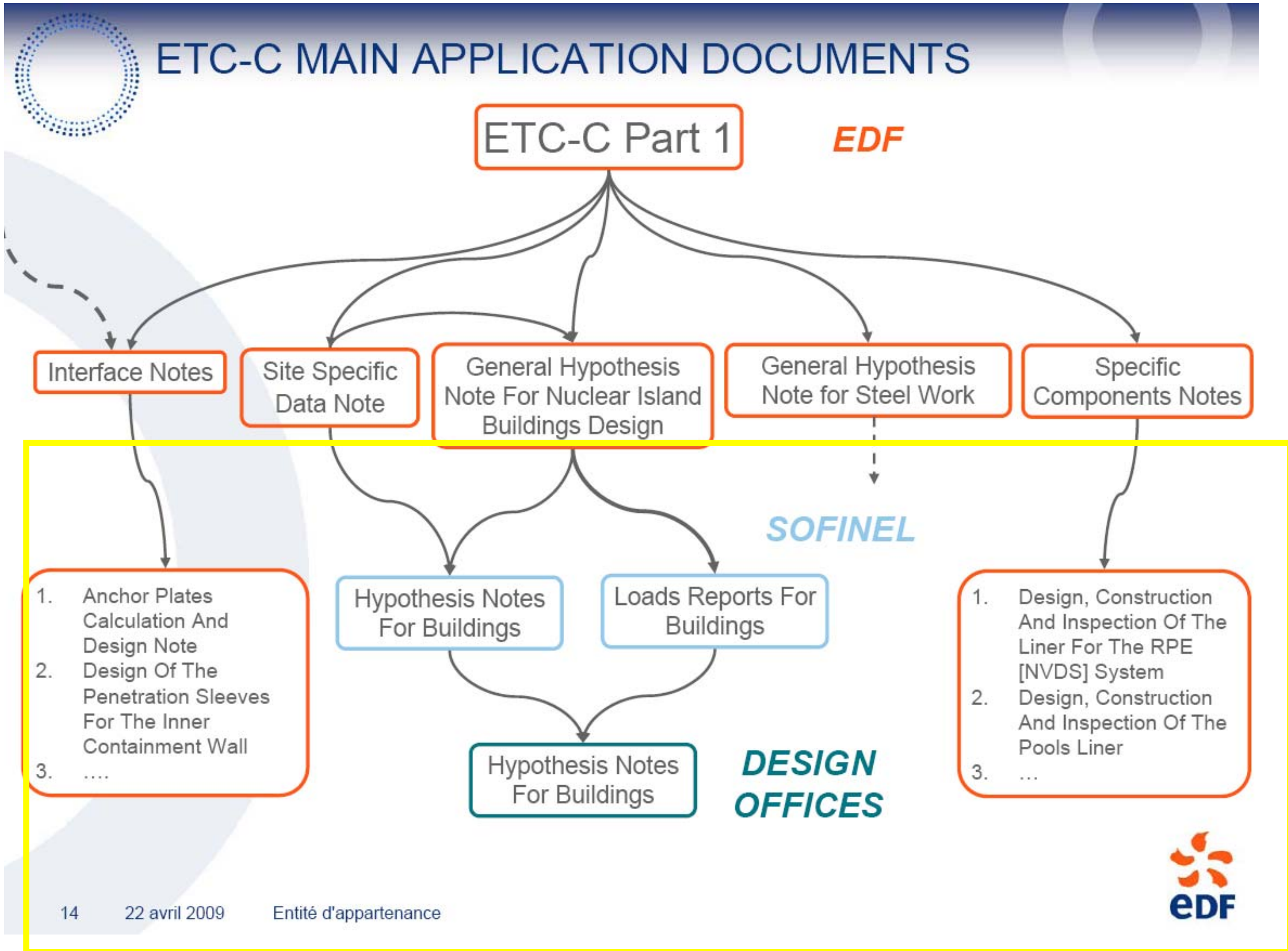


Figure 3: Mind Map of Safety Assessment Principles

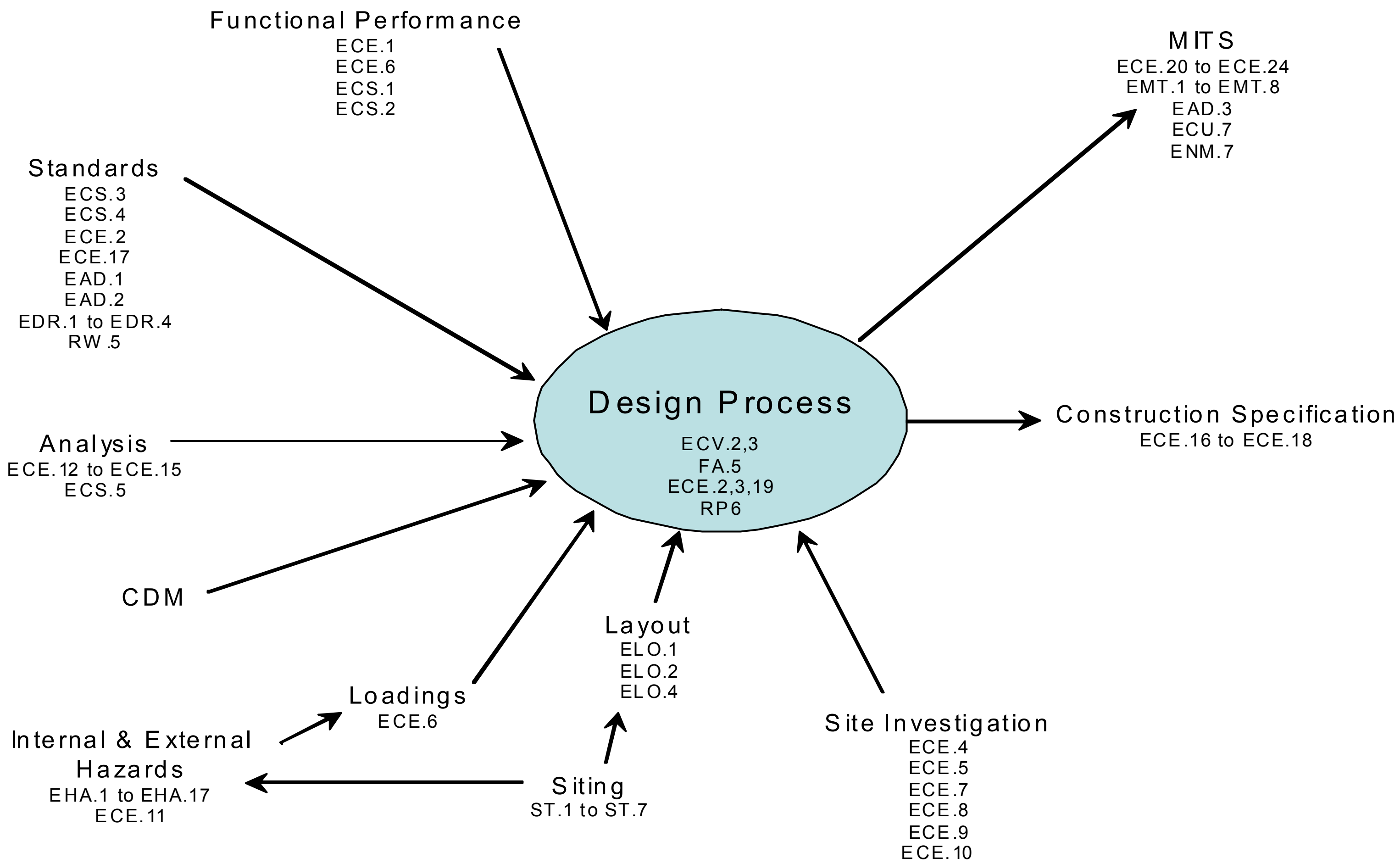


Figure 4: Map Showing Finite Element Code Coverage in the UK EPR Design

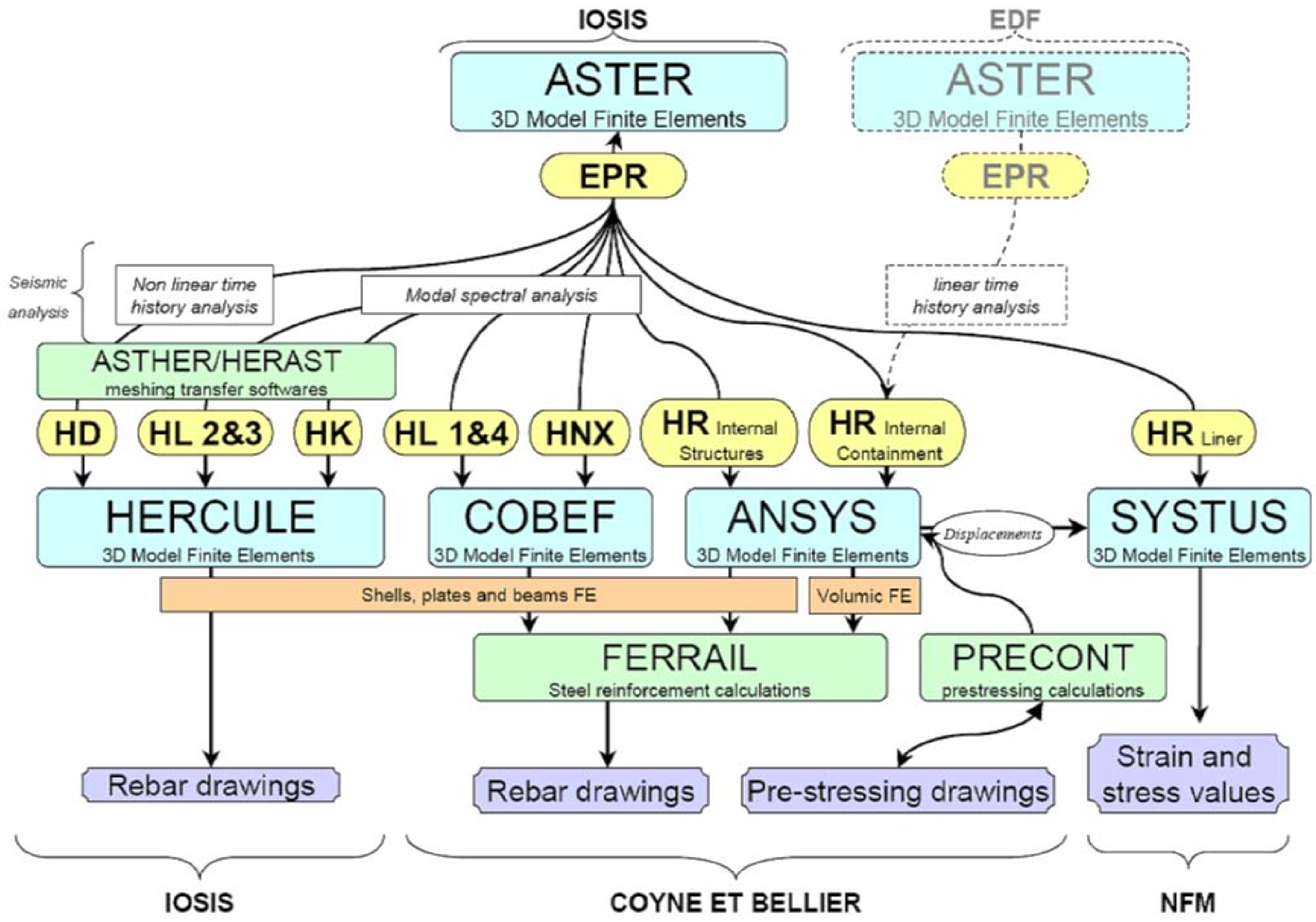
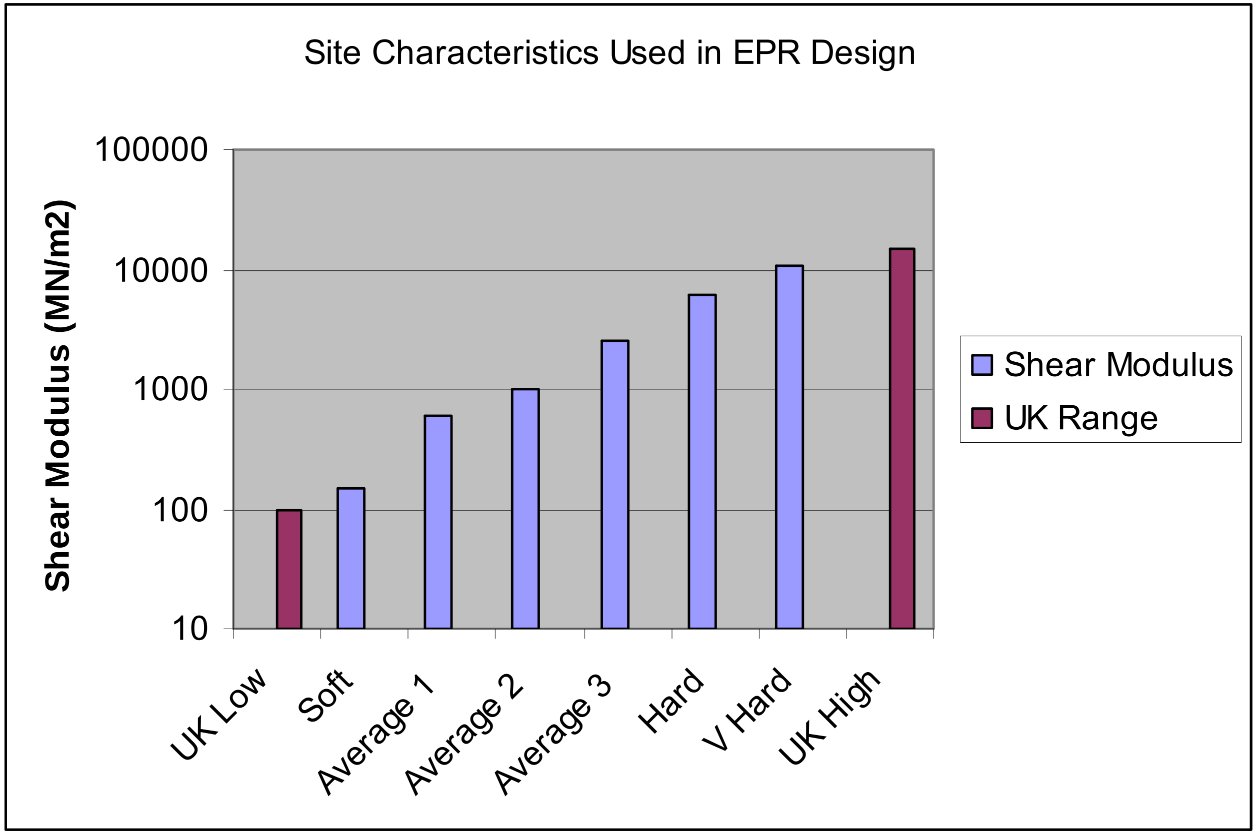


Figure 5: Site Soil Characteristics Used in the EPR design and UK Envelope



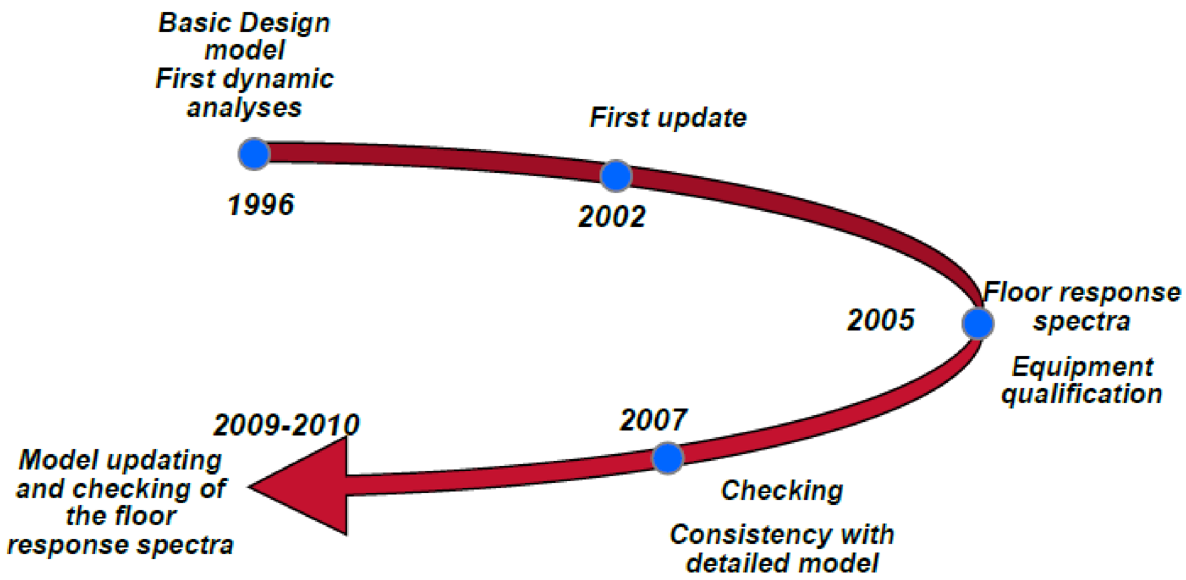


Figure 6: Evolution of the Code ASTER Global NI Model

Figure 7: Example Interface Map for the SAB 1 and 4 Structures (Software Codes and Organisations)

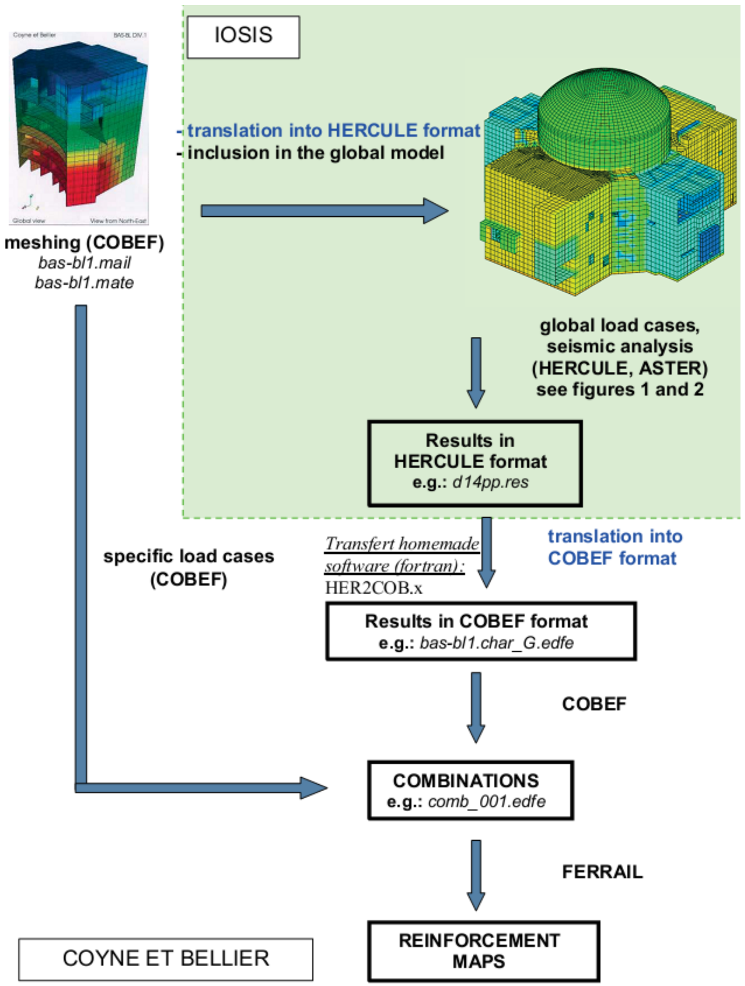


Table 11: Notes on the Design Process Flow Chart

Ref	Note
1	[redacted]
2	[redacted]
3	[redacted]
4	[redacted]
5	[redacted]
6	[redacted] 1. [redacted] 2. [redacted] 3. [redacted] 4. [redacted] 5. [redacted]
7	[redacted] 1. [redacted] 2. [redacted] 3. [redacted] 4. [redacted] [redacted]
8	[redacted]
9	[redacted]
10	[redacted]
11	[redacted] 1. [redacted] 2. [redacted] 3. [redacted] [redacted]
12	[redacted]
13	[redacted]
14	[redacted]

Figure 8: Design Process Flow for the Fuel Building

[redacted]

Annex 1

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

Civil Engineering and External Hazards – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-CE-001	The licensee shall examine the potential for EMI, Industrial hazards, transport threats, fire and release of chemical/toxic material from adjacent sites once a site has been chosen.	First structural concrete
AF-UKEPR-CE-002	The licensee shall derive hazard magnitudes on a site specific basis for those hazards screened out as only capable of evaluation on a site specific basis, including rainfall, flooding, biological fouling and infestation.	First structural concrete
AF-UKEPR-CE-003	The licensee shall confirm that the magnitude of all external hazards considered generically envelope those for the particular site under consideration	First structural concrete
AF-UKEPR-CE-004	The licensee shall confirm that for any structure designed using generic site data that that data is enveloped for the particular site under consideration.	First structural concrete
AF-UKEPR-CE-005	The licensee shall take account of any implications of the outcomes of the Internal Hazards GDA Issues which could affect the design of civil structures.	First structural concrete
AF-UKEPR-CE-006	The licensee Shall undertake any necessary fire tests on reinforced concrete walls using the actual materials to be used in the construction in accordance with the requirements of EN1992-1.2.	Nuclear Island safety-related concrete
AF-UKEPR-CE-007	The licensee shall re-visit all calculations undertaken to ETC-C Rev B which are to be used directly to support the design of the UK EPR to confirm that they are compliant against ETC-C AFCEN 2010 and the UK companion document.	First structural concrete
AF-UKEPR-CE-008	The licensee shall justify the use of MISS3D if used as a design tool for UK sites. Justification of its use and the completion of additional studies may be required.	First structural concrete

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

Civil Engineering and External Hazards – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-CE-009	The licensee shall undertake a review of the implications of the discrepancies in the COBEF qualification documents between the reference solution and the COBEF result. Before structures are constructed which have used COBEF in their design, the impact of this review should be evaluated and any necessary design changes incorporated.	First structural concrete
AF-UKEPR-CE-010	The licensee shall demonstrate that adequate verification and validation of any results from the use of HER2COB is undertaken.	First structural concrete
AF-UKEPR-CE-011	Not used.	
AF-UKEPR-CE-012	The licensee shall ensure that whenever cast in concrete drains with secondary containment are used, they shall be capable of being pressure tested (Inner and outer boundary).	Nuclear Island safety-related concrete
AF-UKEPR-CE-013	The licensee shall provide further details of the earthing and Faraday Cage system such that the impact on the reinforcement bars used can be established. Where welding of reinforcement is proposed, this should be justified.	Nuclear Island safety-related concrete
AF-UKEPR-CE-014	The licensee will need to provide details of the movement joints between the Nuclear Island and adjacent structures in terms of their effectiveness, practicability and longevity.	Nuclear Island safety-related concrete
AF-UKEPR-CE-015	The licensee shall provide details of the waterproof membrane for safety critical structures in terms of its effectiveness, practicability and longevity.	Nuclear Island safety-related concrete
AF-UKEPR-CE-016	The licensee shall develop a hypothesis note for the design of class 1 and 2 steel structures ahead of any site specific design.	Install polar crane
AF-UKEPR-CE-017	The licensee shall develop a hypothesis note for the pool liner design at the site specific stage.	Nuclear Island safety-related concrete

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

Civil Engineering and External Hazards – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-CE-018	The licensee shall develop suitable acceptance criteria for the testing of the pools at the site specific stage.	Install polar crane
AF-UKEPR-CE-019	The licensee shall provide the static soil analysis methodology and results for the Class 1 and 2 civil structures.	First structural concrete
AF-UKEPR-CE-020	The licensee shall ensure that due regard is taken of the effects of Structure- Soil Structure Interaction in the seismic analysis of the Class 1 and 2 structures.	First structural concrete
AF-UKEPR-CE-021	The licensee shall ensure that any seismic analysis undertaken by Code ASTER takes account of missing mass in an appropriate manner.	First structural concrete
AF-UKEPR-CE-022	The licensee shall undertake detailed structural design of the SAB building and provide suitable justifications for the structural forms and reinforcement.	Nuclear Island safety-related concrete
AF-UKEPR-CE-023	The licensee shall ensure that wherever reinforcement areas have been added directly to produce a composite quantity, complimentary checks are undertaken to ensure that this approach is conservative.	Nuclear Island safety-related concrete
AF-UKEPR-CE-024	The licensee shall ensure that wherever ETC-C loadcases are dismissed from the design process that a rationale for this is provided.	Nuclear Island safety-related concrete
AF-UKEPR-CE-025	The licensee will need to ensure that for the fuel building, the simplifications in the mesh used for the structure and the applications of loads are systematically reviewed and justified.	Nuclear Island safety-related concrete
AF-UKEPR-CE-026	The licensee shall review all supplier provided data used in the design of civil structures and confirm their acceptance of it as suitable for that purpose.	Nuclear Island safety-related concrete

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

Civil Engineering and External Hazards – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-CE-027	The licensee shall update the finite element models of Class 1 and 2 structures will to take into account the final geometrical changes applied to the buildings (i.e. based on final layout detailed design).	Nuclear Island safety-related concrete
AF-UKEPR-CE-028	The licensee shall confirm that the concrete portion of all steel lined concrete pools which have a permanent and potentially contaminated fluid shall be confirmed as adequate against the requirements of BS-EN 1992 part 3 (Tightness class 1).	Nuclear Island safety-related concrete
AF-UKEPR-CE-029	The licensee shall demonstrate the suitability of the equivalent lateral load method for the application of seismic loads to Seismic Class 1 and 2 structures if this approach is used.	Nuclear Island safety-related concrete
AF-UKEPR-CE-030	The licensee shall demonstrate the stability of the NAB in terms of sliding and overturning under seismic loading.	Nuclear Island safety-related concrete
AF-UKEPR-CE-031	The licensee will need to undertake detailed structural design of the NAB building and provide suitable justifications for the structural forms and reinforcement.	Nuclear Island safety-related concrete
AF-UKEPR-CE-032	The licensee shall develop a detailed grout specification for use at a specific sites.	Install polar crane
AF-UKEPR-CE-033	The licensee shall develop a robust set of specifications for the grouting mock ups and the acceptance criteria.	Install polar crane
AF-UKEPR-CE-034	The licensee shall develop tensioning and grouting timing restrictions on a site specific basis.	Install polar crane
AF-UKEPR-CE-035	The licensee shall investigate the effects of bursting forces on the integrity of the containment wall and the effects of subsequent strain predictions (if any) incorporated into the proposed layout of containment instrumentation.	Nuclear Island safety-related concrete

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

Civil Engineering and External Hazards – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-CE-036	The licensee shall develop the arrangements for the monitoring of the inner containment.	Nuclear Island safety-related concrete
AF-UKEPR-CE-037	The licensee shall develop the layout of the containment instruments sufficient to satisfy the requirements of the PCSR.	Nuclear Island safety-related concrete
AF-UKEPR-CE-038	The licensee shall develop the test criteria and related monitoring and alert arrangements for the initial and decennial pressure tests on the containment.	Containment Pressure test
AF-UKEPR-CE-039	The licensee shall investigate the potential for using Accoustic emissions and other non invasive detection systems on the inner containment structure as could be available to them at the time of site development.	Containment Pressure test
AF-UKEPR-CE-040	The licensee shall demonstrate the adequate treatment in the design of the Gusset area of any net tensions which develop.	Nuclear Island safety-related concrete
AF-UKEPR-CE-041	The licensee shall justify the approach of using imposed displacements derived from global models with approximate representation of singularity stiffness as boundaries immediately adjacent to the more detailed FE models of the liner and sleeve without modifying the interacting load distribution.	Nuclear Island safety-related concrete
AF-UKEPR-CE-042	The licensee shall provide detailed information and justification of the transfer of shear loads into the inner containment wall for penetrations.	Nuclear Island safety-related concrete
AF-UKEPR-CE-043	The licensee shall justify all loads on the polar crane bracket, including the dismissal of radial inward loads.	Nuclear Island safety-related concrete
AF-UKEPR-CE-044	The licensee shall undertake a complementary analysis of a polar crane bracket to confirm that the assumed load and strain distribution is reasonable.	Nuclear Island safety-related concrete

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

Civil Engineering and External Hazards – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-CE-045	The licensee shall undertake both group behaviour calculations as well as provide evidence that serviceability limits are complied with for the Polar Crane Anchorages.	Nuclear Island safety-related concrete
AF-UKEPR-CE-046	The licensee shall justify the strut and tie approach for the design of the containment wall adjacent to the polar crane brackets. The existing justification does not fully consider the distribution of loads in 3 dimensions.	Nuclear Island safety-related concrete
AF-UKEPR-CE-047	The licensee shall confirm the loads imposed by the RPV ring on the civil structure and provide appropriate justification for their magnitude.	Nuclear Island safety-related concrete
AF-UKEPR-CE-048	The licensee shall provide a revised set of calculations on the RPV support which provide a more complete justification of the approach adopted.	Nuclear Island safety-related concrete
AF-UKEPR-CE-049	The licensee shall confirm the design basis for the tendon anchorages and confirm adherence to the claimed codes and design standards.	Nuclear Island safety-related concrete
AF-UKEPR-CE-050	The licensee shall undertake site specific analyses of the behaviour of the Nuclear Island under aircraft impact to confirm the in-structure responses are within the GDA envelope.	Install polar crane
AF-UKEPR-CE-051	The licensee shall undertake a probabilistic study of accidental aircraft impact on a site specific basis.	Install polar crane
AF-UKEPR-CE-052	The licensee shall develop a hypothesis note for the Turbine Hall.	First structural concrete
AF-UKEPR-CE-053	The licensee shall ensure that the design of the steel framework of the MCR is undertaken in accordance with the ETC-C.	Install polar crane
AF-UKEPR-CE-054	The licensee shall provide justification of the seismic class of all items of structures systems and components in the MCR.	Mechanical. Electrical and C&I Safety Systems – Before delivery to Site

Annex 1

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

Civil Engineering and External Hazards – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-CE-055	The licensee shall produce a revised version of the NAB Chimney hypothesis note for the UK EPR.	First structural concrete
AF-UKEPR-CE-056	The licensee shall develop a fault schedule incorporating external hazards.	First structural concrete
AF-UKEPR-CE-057	The licensee shall develop a consolidated external hazards safety case.	Fuel load
AF-UKEPR-CE-058	The licensee shall confirm that relay chatter is not a concern for the proposed plant and equipment for a particular site either through elimination of components which exhibit this behaviour or by suitable testing.	Mechanical. Electrical and C&I Safety Systems – Before delivery to Site
AF-UKEPR-CE-059	The licensee shall confirm that the seismic fragilities used are valid for the particular site conditions.	Fuel load
AF-UKEPR-CE-060	The licensee shall develop a more refined set of containment fragilities for site specific application to the PSA.	Fuel load
AF-UKEPR-CE-061	The licensee shall develop a set of arrangements for the qualification of plant and equipment against the demands from Internal and external hazards.	Install polar crane
AF-UKEPR-CE-062	The licensee shall undertake sufficient confirmatory reviews for existing work and secondly and to put in place suitable procedures for control of any new works for site specific activities or for new work undertaken in support of resolution of GDA Issues.	First structural concrete
AF-UKEPR-CE-063	Not used.	
AF-UKEPR-CE-064	The licensee shall have in place arrangements to undertake independent nuclear safety evaluation of the Civil Engineering design of safety critical structures.	First structural concrete

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

Civil Engineering and External Hazards – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-CE-065	The licensee shall ensure that due consideration is given to the CDM regulations for the design of all civil structures (safety and non-safety).	First structural concrete
AF-UKEPR-CE-066	The licensee shall demonstrate that adequate margins beyond the design basis exist for all Class 1 civil structures.	Nuclear Island safety-related concrete
AF-UKEPR-CE-067	The licensee shall provide detailed justification of the design of APC shell doors and openings.	Nuclear Island safety-related concrete
AF-UKEPR-CE-068	The licensee shall undertake analysis of the containment structure to reflect the actual concrete properties used in the construction.	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 <u>during</u> the operational phase. For other Assessment Findings, it is the regulators' expectation that the findings are adequately addressed no later than the milestones indicated above.

Annex 2

GDA Issues – Civil Engineering and External Hazards – UK EPR

EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT

GDA ISSUE

HYPOTHESIS AND METHODOLOGY NOTES FOR CLASS 1 STRUCTURES

GI-UKEPR-CE-01 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-01</td><td>GDA Issue Action Reference</td><td>GI-UKEPR-CE-01.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">The specification, methodology and hypothesis notes for Class 1 civil structures have not been found to be fully adequate for use in the design of the UKEPR.</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">ONR raised concerns over the use of ETC-C as a design code in Step 3 of GDA. One key point raised in the response was that ETC-C needs to be read with the particular hypothesis notes for the building under examination. Hypothesis notes are typically prepared at three levels, the highest level by EDF (CNEN), the second level by Sofinel, and the third and most detailed level by the individual design teams for the building in question.

A revised hypothesis note(s) for the Nuclear Island, Safety Auxiliaries Building, Fuel Building, Nuclear Auxiliaries Building, Reactor Building, and the Diesel Building structures shall be produced.

The following areas of concern need to be addressed in the revised document:
The document should be UK specific including definition of ground conditions, climatic conditions and the structural classification.
The overall design life needs to be clarified.
Extensive references are made to French legislation and decrees as well as standards, which are of no relevance in the UK
The PSAR is constantly referred to.
A number of the key references have been superseded.
The document should reflect the latest position on load drops.
There are details on load combinations and replication of aspects of the ETC-C. This may not fully align with the 2010 version of ETC-C and the UK companion document requirements.
There are no apparent requirements to consider robustness or global stability of the NI structures in accordance with the UK Building regulations part A.
There is no reference to the need to consider the CDM regulations.
The document lacks detail in a number of areas including structural philosophy, analysis methods, interfacing with adjacent structures etc.
The sections on the treatment of earthquakes and foundations are inconsistent with the latest methodologies.
The foundation conditions are limited to those of Flamanville.
The use of an equivalent static load method for seismic cases is suggested, which is out with the requirements of ETC-C.</td></tr>
</table>

Annex 2

EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT

GDA ISSUE

HYPOTHESIS AND METHODOLOGY NOTES FOR CLASS 1 STRUCTURES

GI-UKEPR-CE-01 REVISION 1

Technical Area		**CIVIL ENGINEERING**	
Related Technical Areas		None	
GDA Issue Reference	**GI-UKEPR-CE-01**	**GDA Issue Action Reference**	**GI-UKEPR-CE-01.A1**
	• The guidance on the construction of the finite element models for the structure are very weak without reference to other guidance. • The treatment of APC scenarios is unclear. • It is stated that there is a requirement for the reactor vessel pit to be completely dry, however there is no further guidance on how this should be achieved. • For a number of the accident scenarios, the loading is not clearly defined; references are made to future work-scopes. This is the case for some reactor pit thermal loads, internal missiles, and pipework rupture. • There is no design guidance for the treatment of gaps between the NAB and SAB or Fuel Building. • There are a series of vague statements over the future monitoring of foundation movements and references to "current policy". • The option for using projecting bars (bent down bars) in openings is allowed, this is not a practice which is generally permitted in the UK for Nuclear structures. • There are a large number of references to Règles Fondamentales de Sûreté (RFS) documents for derivation of loads. These have not been benchmarked against the UK expectations. • The document states that long term settlement does not need to be considered, which is seen as a shortfall. • There is no detailed discussion on the need for some floor elements to essentially be leak-tight. With agreement from the Regulator this action may be completed by alternative means.		

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EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT

GDA ISSUE

USE OF ETC-C FOR THE DESIGN AND CONSTRUCTION OF THE UKEPR

GI-UKEPR-CE-02 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-02</td><td>GDA Issue Action Reference</td><td>GI-UKEPR-CE-02.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">There is not yet sufficient justification of the ETC-C AFCEN 2010 version and UK Companion Document to confirm these can be used for the design, construction and testing of the UK EPR civil works structures.</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Support assessment within the following areas by providing adequate responses to any questions arising from assessment by ONR of documents submitted during GDA Step 4 but not reviewed in detail at that time:

• a_{cc} Coefficient
• Load Combination Factors
• Biaxial Stress Limits
• Shear
• Fastenings
• Pre-stressing Participation
• Shrinkage
• Crack Width Control
• Pre-stressing Partial Factor

Provide additional supporting documents on the following areas

• Detailing provisions
• Pool Liner Design
• Drop Load Analysis

With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

Annex 2

EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT
GDA ISSUE
USE OF ETC-C FOR THE DESIGN AND CONSTRUCTION OF THE UKEPR
GI-UKEPR-CE-02 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-02</td><td>GDA Issue Action Reference</td><td>GI-UKEPR-CE-02.A2</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide a revision of the UK companion document which addresses the observations raised on ETC-C AFCEN Part 0 as a result of our assessment, the key points being:
• Lack of independent review of the code.
• References to French standards, with translations not provided
• Loose references to “equivalent standards”
• There are no references to national annexes to some standards such as EN13670

In addition, please provide a mapping document (i.e. updated ETC-C assessment file) which identifies how these points have been dealt with.

With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

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EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT

GDA ISSUE

USE OF ETC-C FOR THE DESIGN AND CONSTRUCTION OF THE UKEPR

GI-UKEPR-CE-02 REVISION 1

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<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-02</td><td>GDA Issue Action Reference</td><td>GI-UKEPR-CE-02.A3</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide a revision of the UK companion document which addresses the observations raised on ETC-C AFCEN Part 1 as a result of our assessment, the key points being:
• Errors in Formulas
• Lack of Clarity/ ambiguity in text
• Inconsistency with other sections of the code
• Inconsistency with UK National annexe
• Lack of guidance to designers on seismic design
• Revisions of supporting documents unclear
• Lack of guidance on choice of Eurocode value when no recommended value is available
• Justification lacking for some revised liner stress limits

In addition, please provide a mapping document (i.e. updated ETC-C assessment file) which identifies how these points have been dealt with.

With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

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EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT

GDA ISSUE

USE OF ETC-C FOR THE DESIGN AND CONSTRUCTION OF THE UKEPR

GI-UKEPR-CE-02 REVISION 1

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<tr><td colspan="2">Related Technical Areas</td><td colspan="2">None</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-UKEPR-CE-02</td><td>GDA Issue Action Reference</td><td>GI-UKEPR-CE-02.A4</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide a revision of the UK companion document which addresses the observations raised ETC-C AFCEN Part 2 as a result of our assessment, the key points being:
• Insufficient information to be the basis of a clear construction specification
• Links to French ministerial standards are of no relevance to the UK
• Clarification of the intention to demonstrate the equivalence of French standards to other national standards
• Provide clarity over the approval of modifications or adaptations to the specification
• Provide clarity over how demonstration of equivalence would be achieved
In addition, please provide a mapping document (i.e. updated ETC-C assessment file) which identifies how these points have been dealt with.
With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

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EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT
GDA ISSUE
BEYOND DESIGN BASIS BEHAVIOUR OF THE CONTAINMENT
GI-UKEPR-CE-03 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">PSA</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-UKEPR-CE-03</td><td>GDA Issue Action Reference</td><td>GI-UKEPR-CE-03.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">There is not yet sufficient justification of the beyond design basis behaviour of the EPR containment structure.</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Support assessment of the beyond design basis analysis approach by providing adequate responses to any questions arising from assessment by ONR of documents submitted during GDA Step 4 but not reviewed in detail at that time.

Based on a high level review of the documents and assurances provided to date I have sufficient confidence in the design process to conclude that it should be possible to provide a suitable demonstration of the beyond design basis performance.</td></tr>
</table>

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EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT

GDA ISSUE

BEYOND DESIGN BASIS BEHAVIOUR OF THE CONTAINMENT

GI-UKEPR-CE-03 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-03</td><td>GDA Issue Action Reference</td><td>GI-UKEPR-CE-03.A2</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide a justification of the approach used for the development of the containment fragilities used in the PSA analysis by comparison with the approaches used for beyond design basis assessment.
Based on a high level review of the documents and assurances provided to date I have sufficient confidence in the design process to conclude that it should be possible to provide a suitable demonstration of the containment fragility.
With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

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EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT

GDA ISSUE

CONTAINMENT ANALYSIS

GI-UKEPR-CE-04 REVISION 1

<table>
<tr><td>Technical Area</td><td colspan="3">CIVIL ENGINEERING</td></tr>
<tr><td>Related Technical Areas</td><td colspan="3">None</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-UKEPR-CE-04</td><td>GDA Issue Action Reference</td><td>GI-UKEPR-CE-04.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">The analysis of the UK EPR containment structure has not been demonstrated to capture the behaviour in a sufficiently accurate manner.</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 documents submitted during GDA Step 4 but not reviewed in detail at that time.

During the Step 4 assessment, the following areas were highlighted as requiring further justification:

• Inner Containment seismic calculations in relation with ETC-C requirements.
• Damping ratio of the pre-stressed concrete containment structure.
• Comparison Between Equivalent Static Seismic Analysis of the Pre-stressed Inner Containment and Seismic Spectrum Analysis with Global NI Model
• Simplifications over the representation of the foundation

The combined rationale for the analysis methodology and associated design basis is insufficient in providing a coherent description of the overall analytical process, and fails to adequately address specific analytical aspects necessary to demonstrate a level of structural performance and reliability commensurate with that expected for inner containment.

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.

With agreement from the Regulator this action may be completed by alternative means.</td></tr>
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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>
</table>

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EDF AND AREVA UK EPR GENERIC DESIGN ASSESSMENT

GDA ISSUE

SEISMIC ANALYSIS METHODOLOGY

GI-UKEPR-CE-06 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-06</td><td>GDA Issue Action Reference</td><td>GI-UKEPR-CE-06.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">There is not yet sufficient justification of the methodologies proposed for the seismic analysis of the UK EPR and treatment of the design of the Raft foundation.</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Support assessment associated with the methodology for the seismic analysis of the raft foundation and nuclear island superstructures and provide adequate responses to any questions arising from the assessment by ONR of documents submitted during GDA Step 4 but not reviewed in detail at that time

Based on my full assessment of the earlier reports and a limited, high level review of the reports recently received, I have sufficient confidence in the approach to conclude that it should be possible to provide a suitable demonstration of the suitability of the methodologies proposed.

With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>