Health and Safety Executive

NUCLEAR DIRECTORATE

GENERIC DESIGN ASSESSMENT – NEW CIVIL REACTOR BUILD

STEP 3 CIVIL ENGINEERING AND EXTERNAL HAZARDS ASSESSMENT OF THE EDF AND AREVA UK EPR

DIVISION 6 ASSESSMENT REPORT NO. AR 09/039-P

HSE Nuclear Directorate
Redgrave Court
Merton Road
Bootle
Merseyside L20 7HS

EXECUTIVE SUMMARY

This reports presents the findings of the Civil Engineering and External Hazards Assessment of the EDF and AREVA UK EPR Pre-Construction Safety Report (PCSR) (Ref. 1) undertaken as part of Step 3 of the Health and Safety Executive's (HSE) Generic Design Assessment (GDA) process.

This civil engineering and external hazards assessment report for the UK EPR provides an overview of the safety case in the form of the PCSR as produced by EDF and AREVA, the standards and criteria adopted in the assessment undertaken by ND and an assessment of the claims, arguments and evidence provided within the safety case based upon those standards and criteria.

It should be noted that the Nuclear Directorate (ND) are currently part way through the GDA process and the intent of this Step 3 assessment is to provide an interim position statement regarding the assessment currently being undertaken. There are a number of areas where further detailed assessment is ongoing. In addition, there are a number of areas where we are still awaiting further clarifications and information to allow us to proceed with our assessment.

This report has taken into consideration the findings of the Step 2 Civil Engineering and external hazards assessment of the UK EPR (Ref. 3) and has confirmed that the issues contained therein have been addressed within Step 3 or are intended for resolution within Step 4.

One further complication for these topic areas is the site dependent nature of both the magnitude of the external hazards or the local conditions which may dictate design choices. As a result, there are a large number of areas where definitive statements over the acceptability of the design cannot be confirmed until Phase 2.

The analysis and design of the civil structures has been undertaken using primarily French or European codes and standards, about which we knew little at the start of Step 3. There has been a considerable learning curve therefore before substantive assessment could commence.

The development of the design basis load cases within the documentation has taken some unravelling, however I consider that 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.

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.

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.

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 bonded prestressing 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.

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.

Two Regulatory Observations / Regulatory Observation Actions (RO / ROA) have been raised to which ND has not received satisfactory responses as yet.

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.

In summary, there are a number of areas of further detailed assessment required to be undertaken during Step 4 to provide ND with confidence that an adequate safety case can be made for the construction and operation of the EDF and AREVA UK EPR within the UK and within the UK Regulatory regime.

LIST OF ABBREVIATIONS

ASN	Autorité de Sûreté Nucléaire (French Nuclear Regulator)
ALARP	As Low As Reasonably Practicable
BMS	(Nuclear Directorate) Business Management System
C&I	Control and Instrumentation
DE	Design Earthquake
EA	The Environment Agency
EDF and AREVA	Electricité de France SA and AREVA NP SAS
ETC-C	EPR Technical Code Civil
ETC-F	EPR Technical Code Fire
EUR	European Utilities Requirements
FE	Finite Element
FL3	Flamanville 3 Nuclear Power plant
GDA	Generic Design Assessment
HCLPF	High Confidence of Low Probability of Failure
HSE	The Health and Safety Executive
IAEA	The International Atomic Energy Agency
LOCA	Loss of Cooling Accident
NCB	Non Classified Building
ND	The (HSE) Nuclear Directorate
OL3	Olkiluoto 3 Nuclear Power Plant
PCER	Pre-construction Environment Report
PCSR	Pre-construction Safety Report
PSA	Probabilistic Safety Assessment
RCC-E	Design and Conception Rules for Electrical Equipment of Nuclear islands
RI	Regulatory Issue
RIA	Regulatory Issue Action
RO	Regulatory Observation
ROA	Regulatory Observation Action
RP	Requesting Party
SAP	Safety Assessment Principle
SLLOCA	Surge Line Loss of Cooling Accident
SSC	System, Structure and Component
STUK	Säteilyturvakeskus (Finnish Nuclear Regulator)
TAG	(Nuclear Directorate) Technical Assessment Guide
TQ	Technical Query

LIST OF ABBREVIATIONS

UKEPR	United Kingdom European Pressurised Reactor
WENRA	The Western European Nuclear Regulators' Association

TABLE OF CONTENTS

List of Tables

List of Figures

Annexes

1 INTRODUCTION

1 This assessment report records the Step 3 Civil Engineering and External Hazards Assessment of the Electricité de France SA and AREVA NP SAS (EDF and AREVA) United Kingdom European Pressurised Reactor (UKEPR) submission in accordance with the strategy outlined in Ref. 2.

2 OVERALL STRATEGY

2 This is the second major report on the assessment of the civil engineering and external hazards aspects of the design of the UKEPR. It presents a snapshot of progress at the time of writing, and as can be seen, there are a large number of ongoing areas of assessment.

3 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.

4 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 Step 3.

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

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

7 The reference design is that adopted for Flamanville 3. 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.

3 STEP 2 FINDINGS

8 The Step 2 Siting, Civil Engineering and External Hazards Assessment of the EDF and AREVA UK EPR was reported in Ref. 3.

9 Overall, it was concluded that the EDF and AREVA claims against the key Siting, Civil Engineering and External Hazard Safety Assessment Principles (SAPs) used for Step 2, were reasonable. However, supporting arguments and evidence would be required, during Steps 3 & 4, to ensure that the UK EPR design complies with the claims and also complies, where reasonably practicable, with the full range of Siting, Civil Engineering and External Hazard SAPS.

10 In preparation for Step 3 the assessment made a number of observations which identified further information to be provided by EDF and AREVA in support of the claims.

3.1 Observations Made in Step 2

11 The observations made in Step 2 are repeated below. For each of these observations, work activities have been started in Step 3 or are scheduled for Step 4. Reference is made next to each of the observations to where further information can be found in this report.

1 The design criteria have been clearly laid out, however there is no attempt to rationalise the application to the UK, either by inclusion or exclusion of areas / sites (see Chapter 4.3.1).

2 The grouted duct design for the containment building is not an approach which has been accepted in the UK. Removal of tendons to allow routine inspection, and tightness checks is something which has become standard practice in the UK (see Chapter 4.3.7).

3 The links from design classification to design standards will need further investigation to ensure that the intent is satisfied. Clarity over the design classification for structures will need to be provided (see Chapter 4.3.2).

4 The standards used need to be understood better especially those which appear to be EPR specific. This primarily relates to ETC-C. It is noted that reference is made to principles in Eurocodes, noting that Eurocodes are specifically ruled out as non-nuclear codes (see Chapter 4.3.4).

5 There needs to be a recognition that non French specification materials will be used for construction (see Chapter 4.3.4. and will be addressed as part of Site Licensing).

6 The process for Hazards Identification, definition and consideration of consequential effects will require greater scrutiny in Step 3. The definitions of coincident plant states with hazards will also be reviewed in detail (See Chapters 4.3.1 and 4.3.3)

7 The process of load schedule development will require greater scrutiny in Step 3 (see Chapter 4.3.3).

8 A more considered view of the claims against SAP ESS.18 (“no external hazard should disable a safety system”), including the link to the PRA will be required. This will also include a review of “Cliff edge” considerations. (This will be addressed as part of Step 4).

9 There needs to be a recognition that the Construction Design and Management Regulations 2007 will apply to this project (see Chapter 4.9).

4 STEP 3 ASSESSMENT

4.1 Requesting Party's Case

12 The primary document which presents the Requesting Party's (RP) case is the Pre-Construction Safety Report (PCSR), Ref. 1. The key elements of the PCSR in so far as they relate to the areas of Civil Engineering and External Hazards are presented below.

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

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

14 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.

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

16 The civil structures for the EPR are claimed to have two main functions:

- protecting systems inside the buildings;
- providing a barrier function.

The first function is to protect systems from external hazards, the aim being to prevent consequential internal events. The second function is linked to the mitigation of the radiological consequences of potential failures inside buildings.

17 The main classification for safety related structures are safety class C1 and seismic class SC1. The definitions are provided below.

Safety Class C1

A building is classified C1 if it houses or supports:

- either equipment which fulfils F1 functions;
- or components liable to contain radioactive materials, therefore classified mechanical M1, M2 or M3.

Other buildings are not safety classified: NCB.

Seismic Class SC1

Seismic Class 1 (SC1) equipment and structures as well as related requirements are discussed below.

- Equipment which fulfils F1 functions or is M1-classified, and C1-classified buildings must be Seismic Class 1.
- M2- or M3-classified components may be classified as Seismic Class 1 on a case by case basis in the light of the containment function functional analysis, and taking building requirements into account.
- Generally speaking, systems which fulfil F2 functions need not be SC1. However, the following F2 functions are classified in Seismic Class 1:
    - ultimate diesel generator sets (diesel generator sets for loss of off-site power) and their support systems;
    - the containment heat removal function in the RRC-B condition;
    - partitions, detection and fire-fighting systems must be SC1-classified in the buildings where mechanical, electrical or C&I equipment required for an F1 function is installed;
    - certain additional functions that are not essential to maintain a safe shutdown state but which may be required in the period between 24 and 72 hours post trip.

Seismic Class SC2

Equipment and structures which have to protect or can have an unacceptable impact on seismic class 1 equipment are seismic class 2. Unacceptable impact may result from the following internal hazards subsequent to an earthquake:

- Toppling or falling on to seismic class 1 equipment
- Missiles
- Effects caused by high energy component failures
- Flooding caused by failure in piping and tanks and reservoirs
- Explosion
- Fire

Equipment and structures which do not belong to seismic classes 1 or 2 are referred to as non-seismic classified

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.

18 The classification scheme is summarised in Table 7 of PCSR Chapter 3.2. This table is generally informative, but there is some uncertainty in some of the entries which are open to interpretation. The overall position however can be summarised as follows.

Table 1: 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 Stack	NC	SC2	N	
Fuel Building	C1	SC1	Y (via APS)	
Waste Treatment Building	C1	SC1	N	
Diesel Building	C1	SC1	N	
CW Pumphouse	C1	SC1	Y	
Turbine Hall	NC	SC2	N	
Tunnels/ Galleries	C1	SC1	N	(Buried beneath other structures)

19 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.

20 The second tranche of structures will not be considered in detail within Step 3 or 4 of GDA.

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

22 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 on the common raft.

23 The 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

24 The Reactor Building is made up of a double-walled containment 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.

25 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.

26 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

27 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 and its connected instrumentation & control systems are installed in two of the divisions (2 and 3).

28 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. These divisions include the associated SIS rooms and the control room.
- 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.

29 The Nuclear Auxiliary Building does not house systems or equipment needed to perform F1-classified functions. However, it contains 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 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.

30 The waste 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 waste 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

31 The 4 diesel generators are installed in 2 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.

32 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, and little effort will be expended on examining it during GDA.

33 There are tunnels which run between the various buildings which contain F1-classified systems. Their geographical location ensures that they meet criteria for protection against common mode failure with respect to externally generated hazards, particularly aircraft crash, earthquake and flooding.

4.2 Assessment Strategy

34 The objective of the Step 3 & 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. 4). Ref. 4 was reviewed to produce a SAPs subset for Step 3. In considering the SAPs to be addressed (selection and sampling) during Step 3, the guidance contained in Refs 5 and 6 was followed, for example:

- The Health and Safety Executive HSE) Nuclear Directorate (ND) 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 1.

35 In order to ensure an adequate set of SAPs for Step 3 and 4 a further review of the WENRA reference levels (Ref. 7) and IAEA Nuclear Power Plant (NPP) Design Requirements (Ref. 8) was undertaken. The SAPs selected for assessment of claims and arguments during Step 3 & 4 are shown in Annex 2 of Ref. 2 where they are ordered under assessment topic areas. This is repeated as Table 2 in this document. It should be noted that the number of SAPs to be addressed during Step 3 & 4 has approximately doubled by comparison to those addressed at Step 2.

36 Assessment during Step 3 has tried to address the adequacy of the arguments supporting the claims identified in Step 2. Step 4 will then focus on the evidence to support those arguments. It is difficult to be absolute in the dividing line between arguments and evidence and in some areas the Step 3 assessment has clearly advanced into Step 4.

37 The Step 2 assessment revealed that the design implementation was largely incomplete. The GDA Guidance to RPs (Ref. 5) states for Step 3 *"The requesting party is required to: Provide a detailed Pre Construction Safety Report that includes the following:*

"3.13 confirmation of:

(a) Which design aspects and its supporting documentation are complete and are to be covered by the Design Acceptance Confirmation,

(b) Which aspects are still under development and identification of outstanding confirmatory work that will be addressed during Step 4."

38 The above allows EDF and AREVA to exclude design aspects from the GDA Design Acceptance Confirmation. The GDA process more generally allows for such exclusions (e.g. Ref. 5 para. 62) by the use of formal Exclusions and Commitments, however HSE will seek to minimise any such requirements.

39 Technical Support Contractor(s) (TSC) have been engaged to assist with the assessment work. Section 4.6 provides further details.

40 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. Section 3.17 of Ref. 5 states that *"Reviewing what overseas regulators have done and how HSE can make use of it"* will be undertaken in Step 3. Section 4.7 contains more information.

41 Finally, in summary, the key activities undertaken during Step 3 are as follows.

i) Assessment of responses to Step 2 observations.

ii) Set up and management of TSC support.

iii) Assessment of the claims and arguments for Step 3.

iv) Definition of the scope of items to be considered further in Step 4.

v) Identification and management of relevant GDA related research.

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

42 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 Step 4 is also detailed in the following paragraphs.

4.2.1.1 Design Classification and Load Schedule

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

- 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?

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

4.2.1.2 Codes

45 Within Step 3 the codes are being 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?

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

4.2.1.3 Analysis

47 Within Step 3 the analysis tools are being 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 include:

- 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?

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

4.2.1.4 Implementation

49 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 has been undertaken. It has been primarily restricted to a review of the operational feedback from the two current EPRs under construction at Flamanville and Olkiluoto. However, a more detailed review of the ability of the prestressing elements of the containment to be constructed, particularly the grouting operations has been undertaken in Step 3. 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.

4.2.2 Inclusions

50 The key physical areas of inclusion in the Step 3 review are

- Design of the Containment structure
- Design of the Nuclear Island

4.2.3 Exclusions

51 At this stage the following aspects have not been reviewed in any detail

- Design of the Fuel Building (Not Reviewed as yet)

• Design of the Waste Treatment Building	(Design not developed)
• Design of the Cooling Water Pumphouse	(Site specific)
• Design of the Ancillary Buildings	(Site Specific)
• Design of Ancillary services and structures, i.e. tanks, service trenches	(Site Specific)
• Decommissioning Claims	(Insufficient Information)
• Detailed Review of Aircraft Impact	(Information Delay)
• Review of the Core Catcher	(Step 4 activity)

52 In addition, those aspects of the design which are clearly site specific such as the derivation of external hazard magnitude values for the site have not been considered.

4.2.4 Standards

53 During Step 3 our assessment of the proposed design is against those principles in the HSE Safety Assessment Principles for Nuclear Facilities (SAPs),that are deemed relevant to system design aspects (see guidance below).

54 With regard to the WENRA Reference Levels, 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.

55 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 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.

56 The use of the SAPs is supplemented, as appropriate, with NII Technical Assessment Guides (TAGs). The TAGs 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 the RPs (SAP ECS.3 'Standards'). Consequently, particular attention has been paid to such claims (e.g. has the RP claimed adequate standards selected and applied).

57 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.

- We have been selective in our confirmation of SAPs coverage in Step 3, 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 3 and the level of detail to which the assessment will be taken. The focus has been on those elements with the potentially greatest effect in risk reduction in addition to any elements employing novel or complex techniques.

58 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 3, 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.
- The IAEA document 'Safety of Nuclear Power Plants: Design – Requirements – NS-R-1' supports the selection of the SAP.

4.2.5 Design Completeness

59 The design as assessed in the areas of Civil Engineering and External Hazards is heavily based on the design for Flamanville, which is under construction at present. In practical terms this means that for the Nuclear Island structures all the key building dimensions are fixed, the major member sizes are fixed, however the level of detailed design to include secondary structures (Stairways platforms etc) is not complete. The design of the principle reinforcement for the concrete structures and the prestressing system is essentially complete. Details around penetrations which are required for construction, closers, lacers, and local adjustments for embedments are not complete at this stage. Indeed they are being developed as part of the construction programme at Flamanville.

60 The design as applied at Olkiluoto is not considered as representative of the reference design for GDA. Some information from the construction programme and the regulatory review has been examined as part of the background information for this report however.

4.3 Assessment Progress

61 The following subsections detail the progress to date on each of the key technical areas being assessed. Each section starts with an overview of the scope of the assessment, and an indication of the key SAPs which have informed the review thus far. The next sections focus on a description of the approach by EDF and AREVA and our findings thus far in the assessment.

4.3.1 Generic Site Envelope

62 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.

4.3.1.1 Scope

63 The 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.

64 The key Steps being undertaken during Step 3 and 4 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
  - 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

4.3.1.2 Standards

65 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

66 More useful guidance can be found in the section of the SAPs on External Hazards (EHA.1 to EHA.16). T/AST/013 ‘External Hazards’ (Ref. 10) provides more detailed information on regulation of external hazards.

67 IAEA Documents NS-G-1.5, 1.6 and NS-R-3 provide additional guidance (Refs 11, 12 and 13).

4.3.1.3 Findings – External Hazards

68 Section 2 of the PCSR identifies the generic site that the UKEPR 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

69 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

70 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)

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.

71 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. 14). 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.

72 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.

73 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.

74 It is considered however that the treatment of EMI and release of chemical/ toxic material 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.

75 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.

4.3.1.4 Findings - Site Conditions

76 The UKEPR 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.

4.3.1.5 Summary

77 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 UKEPR design is considered reasonable.

78 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.

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

4.3.2 Design Classification

4.3.2.1 Scope

80 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.

81 The scope of this task covers all buildings which are being considered as part of the GDA (See Section 4.2.2 and 4.2.3).

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

4.3.2.2 Standards

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.		

4.3.2.3 Findings

83 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.

84 Section 4.3.2 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.

85 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. There is an outstanding action for EDF and AREVA to provide further clarification.

86 The response to TQ-EPR-058 has confirmed that Table 7 of Chapter 3.2 of the PCSR *"gives examples of different structures within safety classified buildings being assigned to different classification categories*". This is interpreted that the detailed classification of individual sub-elements of structures may not be resolved until the more detailed hypothesis documents have been produced. However one useful point of clarity is that "the decision has been made to classify every structural member of C1 buildings at SC1 for the Civil Works design."

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

- Main mechanical systems (Sub-Chapter 3.2, Table 3)
- Main electrical systems (Sub-Chapter 3.2, Table 4)
- Main fuel handling and storage systems (Sub-Chapter 3.2, Table 5)
- I&C systems and equipment (Sub-Chapter 3.2, Table 6)
- Civil engineering structures (Sub-Chapter 3.2, Table 7)

88 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 preliminary review of the classification has not given any cause for concern.

4.3.2.4 Summary

89 The classification system is simple to understand, and for the major civil structures the rationale is clear.

90 For some substructures within the main buildings the classification is not fully clear, and will require further investigation as part of Step 4.

4.3.3 Load Schedule

91 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 Chapter 4.3.1). There is, therefore, some overlap with the findings in Section 4.3.1 above.

4.3.3.1 Scope

92 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.

4.3.3.2 Standards

93 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.		

4.3.3.3 Overview of Load Schedule Development

94 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 a preliminary report, Ref. 14.

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

96 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 claim 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. 15).
- 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."

97 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.

98 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 claim 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)".*

99 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

100 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)

4.3.3.4 Design Application

101 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.

102 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 crashes: these are subdivided into three load cases representing general, military and commercial aircraft categories with different time load functions.*
> - *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."*

103 The list of external hazards above is considered reasonable as a basis for design, however the necessary combinations, treatment of consequential effects and beyond design basis considerations require a more detailed consideration.

104 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.

105 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.

106 The design incorporates what is termed an 'Event based approach' Report ENSN040070 (Ref. 16) provides further details.

107 Section 2: of Ref. 16 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)."*

108 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."*

109 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."*

110 This approach is re-stated in Sub-Chapter 13.1 Section 1.3:

> *"In general, the question of combined events may be addressed in three ways:*
>
> - *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."*

During Step 3 we have questioned EDF and AREVA at length on the practical application of these rules, and they continually refer out to Reference :36.

111 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. Further TQs have been raised for clarification.

112 TQ-EPR-056 tries 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."

113 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 2 overleaf.

114 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."*

115 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."

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 2: Summary of the Consequential Hazards Designed for in the EPR

Initiating External Hazard	Consequential Hazard
Earthquake (+LOOP)	Fire
	Internal Explosion
	All remaining Internal Hazards
	External Flooding
	External Explosion
Aircraft Crash	Fire In Buildings
	Fire Outside Buildings
	Internal Flooding
	External Flooding
Industrial Risk (Expolsion)	Internal Flooding
	External Flooding
Wind (+LOOP)	Wind Generated
Lightning	Fire, internal explosion

116 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.

117 Figure 5 shows the 4 levels of documents which have been identified through the work undertaken thus far. The review of these hypothesis documents is ongoing, however several key points have become apparent already.

118 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.

119 As part of the Step 4 review a more detailed sampling of the Level 2,3 and 4 hypothesis documents will be 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.

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

Sub-Chapter 3.3 Section 1.3.5 states that:

> *"For external events, the design of structures must make provision for loadings beyond the design basis, whether they are due to natural phenomena such as earthquakes or climate changes, or to human activity such as explosions or aircraft crashes."*

121 The design also takes into account a double-ended guillotine break of the reactor coolant pressure boundary (LOCA-2A). In addition the combined loading due to a simultaneous loss of coolant accident (reactor coolant system pressurizer surge line break LOCA) with the design earthquake; the purpose of designing against this loading 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.

122 This explains the inclusion of the SLLOCA+DE event in ETC-C Table 1.3.5.1. 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 PSA. The SMA targets a 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 will require closer scrutiny in Step 4.

4.3.3.5 Summary

123 A formal process for identification and screening of external hazards is not evident, the approach being an historical one.

124 The list of external hazards considered in the EPR design appears to be reasonable.

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

126 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.

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

128 ETC-C includes coincident hazard load combinations, such as from wind and snow;

129 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).

130 The 'Event Based Approach' includes consequential hazards, such as fire following earthquake.

131 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.

132 In summary, we consider that the EPR design is likely to meet the requirement of the SAPs. However, the approach adopted has not lent itself to ready scrutiny. Further activity in Step 4 will be required to finalise the assessment in this area and carry it forward into the assessment of how the principles have been applied in practice.

4.3.4 Design Codes

133 Each of the Civil Structures has been designed using the EPR Technical Code-Civil (ETC-C) Ref. 17. 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.

4.3.4.1 Scope

134 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:
  - Application to Nuclear Structures
  - Development History
  - Currency
  - Derogations from standard application
  - Previous regulatory interactions (UK and elsewhere)
  - Code Interfaces
  - Reliability targets
  - Degradation/ lifetime allowances

- Review the application into selected structures:
    - Containment
    - Reactor Buildings
    - Essential Plant Buildings
    - Diesel Buildings
- Review code application in light of above.

4.3.4.2 Standards

135 The key SAPs (and associated guidance where relevant), 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.		

136 Further guidance on the development of codes and standards is also contained within British Standard (BS) 0 'Standard for standards' (Ref. 37) 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.

4.3.4.3 Overview of ETC-C

137 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 & 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 &3
  April 2006 last version

138 The ETC-C is comprised of 3 basic parts

- Part 1: Design
  - i) Actions and combinations of actions
  - ii) Concrete structures (criteria from EC2 + complements)
  - iii) Metal parts contributing to leak-tightness (containment liner and penetrations, pool liners)
  - iv) Steelwork and plate anchorages
  - v) Annexes: seismic analysis, shrinkage & creep, simplified methods for impact (military aircraft) and perforation calculations.
- Part 2: Construction
  - i) Materials: soil, concrete, formwork, rebars, prestressing, precast
  - ii) Penetrations, liners for containment & pools, steelwork
  - iii) Tolerances for procurement, construction
- Part 3: Instrumentation (monitoring) & tests
  - i) Leak-tightness tests

ii) Instrumentation and mechanical resistance tests.

139 **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:

- Safety requirements
- Site data (earthquake, temperature, wind, etc…)
- Special Project requirements
- Specific building actions and rules
- Components covered by specific Technical Specifications such as
  - i) Airlocks/Electrical penetrations
  - ii) Plates
  - iii) Metalwork
  - iv) Paintings
- Relationship with constructors: procedures, checks and controls
- Details of monitoring

4.3.4.4 Findings

140 For Step 3 the primary focus has been on Part 1, although the review of the containment design (See Section 4.3.7) has considered Part 2 in more detail, especially with respect to the grouting activities. In addition, some review of the containment testing operations has been undertaken.

141 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.

142 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.

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

EN1999 Eurocode 9: Design of aluminium structures

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

143 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. Eurocodes 1: Actions on structures – Part 2: Traffic loads on bridges).

144 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 will be followed by a national annex.

145 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.

146 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.

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

148 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"*

149 The requirements within Part A of the building regulations (2004) 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. This will inevitably mean that the Eurocodes with the UK national annexe will be the approved documents. The use of other countries' national annexes is not ruled out; however some degree of parity will be required to be demonstrated.

150 The ETC-C is an EPR specific design code. Our review of the code has confirmed 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. (Under consideration by RO-UKEPR-037)

- There is no evidence of benchmarking against other codes

151 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 are may not be fully compatible at present.

152 I have written to EDF and AREVA (Ref. 18) identifying these issues and a reply is under preparation.

153 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

These issues are being progressed through the route of technical queries. In Step 4, the effects on the design will be examined in more detail.

154 The one key issue which has been 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.

155 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). Following this, discussions were held and a set of actions agreed. In addition, we have held a workshop with a selection of technical support contractors to discuss other options for considering the reliability of the ETC-C code. This has generated a new workstream which is currently ongoing, and expected to report before the end of 2009. This new workstream essentially compares the use of ETC-C on concrete structures via Eurocode 2 and the French National annexes as applied in the generic design at Flamanville and the design that would apply in the UK were UK national annexes to be used. The ensuing reliabilities (as calculated using the Eurocode 0 approach) will then be compared against regulatory expectations.

156 As part of our review of ETC-C, we raised TQ-EPR-147 asking for details of the development path for the code. The response to TQ-EPR-147 consists of EDF Report ENGSGC090215A and associated references. This provides an overview of the development path, peer reviews and the history of engagement with ASN. The following sections provide a commentary on this response.

157 ETC-C has not been developed under the auspices of the AFCEN organisation which is the 'French Association for the Design, Construction and Operating Supervision of the equipment for ElectroNuclear boilers'. It should be noted that the executive committee of AFCEN is entirely populated by EDF and AREVA staff. It is suggested in the response to TQ-EPR-147 that following the use of ETC-C on Flamanville 3, a revised version of ETC-C will be presented to AFCEN. The forerunner of ETC-C, RCC-G was published via AFCEN.

158 The need for developing ETC-C was driven by a number of factors

- More severe load cases than used in RCC-G
- EUR requirements
- Developments in material models
- Changes to French and German Regulatory expectations
- Use of a double containment with a steel liner.

159 There is evidence of some review by the French and German nuclear regulatory bodies, although this appears to be at a safety principles level. Some review of sections of the code has been undertaken by individual experts from either French universities or in-house within EDF or affiliated companies. It is stated in the response to TQ-EPR-147 that these comments were 'largely incorporated', although what this means is unclear. There is no evidence of a wholesale review of the code at a detailed level.

160 There is no evidence that ETC-C has undergone any benchmarking against the previous version of the code (RCC-G) or against other international or national nuclear codes.

161 ETC-C is clearly limited in application to the design of civil nuclear structures on the EPR.

162 It is clear that in order to undertake a design using ETC-C, a large number of associated documents and standards are required, including hypothesis documents, Eurocodes, safety requirements documents and other guidance documents. It is therefore clear that in order to effect a credible design using this code, the designer will need considerable training and support to ensure that its intent is met.

163 The following sections identify those areas where work is ongoing at a detailed level and potential issues have been identified. There are a great number of other smaller issues, the significance of which individually is minor, but will need to be considered in more detail as composite effects in Step 4.

164 The ETC-C approach to shear design is being examined in some detail. The ETC-C introduces an additional method for shear design with links to that given in EN1992. It would appear from an initial examination that higher capacities for lower areas of shear reinforcement are predicted using ETC-C. Shear failure is generally a non ductile one, and the prospect of a lowering in the reliability of the shear design from the normal design route is something which will be investigated thoroughly in Step 4.

165 Section 1.3.3.5.6 of the ETC-C 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 within the ETC-C 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. This is therefore an issue for resolution as part of site licensing activities.

166 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. This will need clarification in Step 4.

167 A further issue on durability is the potential for discrepancy between traditional UK practice in BS 8500 and the cover requirements in ETC-C. Reconciliation of these two approaches will need to be undertaken in Step 4. Additionally, there will be a need for some site specific consideration of exposure classes in Phase 2.

168 It is not clear whether the ETC-C requires crack widths to be checked explicitly but the general indication is that this is not required and reliance is placed on the minimum steel requirements. Given the tensile steel stress limits for the various load cases relatively large cracks could be expected. There is currently insufficient information to look at this in detail but given the effects of crack widths on the integrity of the liner it would seem appropriate to rigorously control their widths. The only place the ETC-C mentions a crack width limit and therefore potentially requires a check is in the design of the galleries and buried pipes under the design earthquake. This will need clarification in Step 4.

169 Sub-section 1.4.5.3 ‘Stress limitations for group 2 situations’ of ETC-C includes the internal accidents, severe accidents and the design earthquake. Again there is an allowance for increased strength due to triaxial effects without a specific check. In addition the concrete design strength on the shear calculation is taken as the characteristic strength. This is not consistent with EN1992 nor with the rest of the document where, for accidental cases, the material safety factor on the concrete is reduced to 1.2, i.e. the design strength should be 0.83 times the characteristic strength. This will need clarification in Step 4

170 ETC-C assumes there are no defects or geometrical deviations in the liner when assessing strains whereas ASME III Division 2 recommends that such imperfections are included. Initial imperfections may well limit the subsequently induced strains but there could be circumstances where strains will be higher if imperfections are considered. Part 2 of ETC-C gives details of the tolerances for the liner, which are reasonably generous. The implications of this need clarification in STEP 4.

171 The strain limits outlined in ETC-C Section 1.5.1.4.2 are generally larger than those suggested in ASME III. The implications of this, particularly for the material concerned, and the quality requirements on the ultimate strain will need to be examined in Step 4.

172 ETC-C states that the liner must not tear if the anchor fails. There is no method, in the document, of assessing this by design or analysis and reference is made to ‘adequate’ tests being performed to show that the stud anchors do indeed fail first. This is something which would need to be examined as part of the site licensing activity.

173 Within Section 1.5 of the ETC-C there is no check for combined tension and shear on the liner stud anchors which would reduce the capacity of the anchor below the individual allowable forces. A combined check is present in Section 1.8.5 of ETC-C (Interface Requirements between Anchors and Concrete) but the Sections are not cross-referenced. This is a general comment on Section 1.5, as the anchor capacities may well be limited by the concrete capacity, a situation not explicitly mentioned in ETC-C but noted in ASME III Division 2. This will need clarification in Step 4.

174 The ETC-C design limits for the containment liner are reported to be justified by experimental testing. It is also noted that the concrete does not seem to influence the ultimate load behaviour of the anchor details used on existing PWR containment buildings. In addition, ETC-C rules are only applicable to the referenced liner and anchor materials. This will need clarification in Step 4

175 There are a number of clarifications in TQs that have been raised on the ETC-C, and more are anticipated. It is not clear how this will be reflected in the application to the

design. For future works, a revision to the standard could capture them. For structures which have already been designed, it is less clear how the importance of these clarifications could be captured.

4.3.4.5 Summary

176 The following key points have emerged from the assessment thus far;

- 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.
- There is a lack of clarity of which Eurocode rules may be used with and without modification.
- There are references to dated design standards (e.g. Eurocodes) and Euronorms throughout.
- A clear statement on the target reliability to be achieved through use of the rules is required. (Under consideration by RO)
- ETC-C does not define a unique set of design rules. There is a need for additional documentation to specify the design to achieve the required level of reliability, the nature of which is not clear.
- There are a number of areas where the detailed technical review is highlighting areas where there are potentially significant deviations from standard UK practice.
- The compatibility between the design rules and the workmanship rules needs further investigation as it appears that they are may not be fully compatible at present.

177 At this stage in our assessment EDF and AREVA have not provided sufficient information to demonstrate that the ETC-C is appropriate for the design of nuclear structures.

4.3.5 Finite Element Codes

4.3.5.1 Scope

178 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.

179 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 are undergoing particular attention 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.

180 The key Steps during the Step 3 and 4 assessment are

- 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. 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.

4.3.5.2 Standards

181 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.		

182 In addition, guidance provided by NAFEMS (Ref. 19) is used for supplementary guidance.

183 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.

184 EDF and AREVA, however, have a different approach to this subject which was clarified in Ref. 20, 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.

185 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.

4.3.5.3 Overview of Codes Used

186 A wide range of Finite Element Codes and related software has been used in the UK EPR design. Figure 2 provides an overview of the most important codes used. The initial phases of Step 3 have focussed on understanding the codes used, and their interrelationship. The sections below summarise the types of code used and their applicability in the design. Following the basic code descriptions, the intended strategy for the assessment of the codes across Step 3 and Step 4 is presented. Following that is a summary of progress to date. It is unfortunate that progress has not been as far as wished. The key reasons for this are the extended time taken to receive documents from EDF and AREVA and the time taken for translation of key documents. It should be noted that there are often extensive document libraries associated with each of the codes. For example, Code ASTER has almost 2000 documents associated with it. As part of the assessment we have focussed on a smaller subset of these (20-30) documents which are central to the operation of the code, and its verification and validation.

187 **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.

188 **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 Elements 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.

189 **Code EUROPLEXUS** (*http://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. Code EUROPLEXUS is used to model the impact of aircraft onto the aircraft protection shell of the nuclear island and the CW pumphouse.

190 **Code COBEF** (*http://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.

191 **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.

192 **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.

193 **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.

194 **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.

195 **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.

196 **Code HERCULE**

Code HERCULE is a general FE code and code checker developed by SOCOTEC in France, and used by IOSIS on some of the Nuclear island structures.

4.3.5.4 Overall Strategy

197 For Step 3:-

Evaluate the key finite element codes used in terms of their capability, reliability for application to the EPR design. This will be at a principles level. This will cover the following codes:

- Code ASTER
- Code ProMISS3D
- Code Europlexus
- Code Systus
- Code Precont
- Code Ferrail

198 For Step 4:-

1 Evaluate the application of the codes identified in Step 3 to the design by selection of key areas for study in terms of analysis and importance within the overall safety of the facility. In addition, examine the application of ANSYS to the containment design.

2 Evaluate the suitability of the assorted other software packages, i.e.

- ASTHER/HERAST
- HERCULE
- COBEF

4.3.5.5 Progress to date

199 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 (Ref. 21). 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. A second meeting which focussed solely on the development and testing of Code ASTER was held on 11th June 2009 (Ref. 22). This provided a better insight into the background management of the codes.

200 **Code ASTER**

The first Steps in the assessment of Code ASTER was to establish which components of the code have been used in terms of element types, loading types and analysis types. A review of Ref. 23 has identified this more clearly. The high level quality assurance documents have been supplied by EDF and AREVA, and all documents relating to the application and qualification of the code have been provided in the original French. A subset of these has been requested in English, some of which have been received and reviewed. The review of Code ASTER is work in progress, however the following impressions have been gained thus far;

- The code is subject to a rigorous QA arrangement which controls versions, access and the verification and validation of the code.

- There are a wide range of verification and validation examples which appear to capture those elements of the code used for the analysis of the civil structures of the EPR.
- There is some evidence of benchmarking against other codes which will be examined as part of Step 4.

201 **Code ProMISS3D**

ProMISS3D is a boundary element code which has been adapted to be linked to Code_Aster. Boundary elements use a mathematical formulation to represent the behaviour of a semi infinite domain of soil or rock. ProMISS3D uses Green's functions to model the soil or rock body around a foundation. It is intended for the analysis of one or multiple foundations and may include fluid interfaces. The soil or rock model may include layers/strata of different properties. As is typical for this class of analysis, the soil or rock strata are treated as linear elastic materials.

An initial review of the verification and validation documentation has been undertaken. This principally compares the results of analyses undertaken with ProMISS3D v2.1 and SASSI 2000. Rather complex problems have been considered. The comparison notes quite large differences in some cases between the results obtained using SASSI and those obtained using ProMISS3D. However, the report does not seek to explain the reason for the observed differences in the analyses or indicate whether the SASSI or the ProMISS3D results are likely to be more reliable

These issues will be examined in more detail as part of the Step 4 assessment.

202 **Code Europlexus**

Limited progress has been made on this code, primarily as a result of the hiatus in access to protectively marked information which details the aircraft impact analysis. The code manuals have been examined, however due to the wide range of options and element types therein, it is not considered appropriate or necessary to examine in detail all aspects of the code. Once the range of element types and analysis is known, a limited review of the code will be undertaken; this will be in Step 4.

203 **Code Systus**

This code was used as part of the Sizewell B project (Ref. 24), for analysis of the reactor pressure vessel, and therefore has some previous pedigree. As a result, there has been no substantive examination of the code at Step 3. A more complete review of its use and capabilities will be undertaken in Step 4.

204 **Code Precont and Ferrail**

Initial discussions were held with Coyne et Bellier in August 2009 (Ref. 25) to gauge the extent and nature of the code and to judge the level of any future regulatory review. A subsequent inspection (Ref. 26) 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. An initial review of the manipulation of the output from finite element analysis suggests that an appropriate approach his adopted in the code.

4.3.5.6 Summary

205 Code ASTER is in the process of being reviewed. At this stage I am confident that it will be found suitable for the analysis of nuclear civil structures under static and seismic loadings.

206 Code ProMISS3D is in the process of being reviewed. The initial review of the verification cases has a shown a larger than expected deviation from the comparison case. Further examination of the code and its documentation will be undertaken in Step 4.

207 The progress on Europlexus is extremely limited, and considerable effort will be required in Step 4.

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

209 The following codes will be examined in terms of their application into the design process during Step 4;

- Code ASTER
- Code ProMISS3D
- Code Europlexus
- Code Systus
- Code Precont
- Code Ferrail

210 The following codes will be examined at a principles and applications level in Step 4.

- ASTHER/HERAST
- HERCULE
- COBEF

4.3.6 Nuclear Island Structures

211 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.

4.3.6.1 Scope

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

4.3.6.2 Standards

213 The key SAPs identified in Sections 4.3.2, 4.3.3 and 4.3.5 are all relevant to the design of the nuclear island structures.

4.3.6.3 Findings

214 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 containment are protected against aircraft impact.

215 The fuel building will contain the fuel storage ponds, the design of which is detailed within the ETC-C. The design approach is primarily based on test results from non destructive and destructive tests on a particular pond liner design. This has already elicited some TQs, and we will be reviewing them further in Step 4.

216 The remainder of the nuclear island structures have not been examined in particular detail as yet. The analysis codes, design codes and load schedules developed have been examined at a generic level, as can be seen in Sections 4.3.2, 4.3.3 and 4.3.5 of this report. Within Step 4 a sampling of the application of the design principles will be undertaken, and a more considered view of the individual structures made.

217 The common foundation raft beneath the nuclear island is a complex structure, the detailed design of which has not been considered as yet. Its assessment is heavily dependant on the examination of the analysis codes, design codes and load schedules, and additionally, the inner containment design. At this stage, definitive statements over the acceptability of any of these individual aspects of the design toolbox cannot be made. As a result, more substantive statements over the raft foundation cannot be made.

4.3.6.4 Summary

218 The design classification of the nuclear island structures is seen to be appropriate given their safety functions 4.

219 Detailed consideration of the structures is contingent on understanding the analysis and design codes in more detail. This will be undertaken as part of Step 4.

4.3.7 Containment Structure

4.3.7.1 Scope

220 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)

221 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

222 The scope covered in Step 3 relates primarily to the civil structure and does not impinge upon the other systems such as the spray systems.

4.3.7.2 Standards

223 The key SAPs (and associated guidance where relevant) which are applicable are as indicated in Section 4.3.6.2 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.		

"424 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."

224 The specialised and particular nature of the containment design with grouted in place prestressing 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. 27). This will also be used to inform my assessment of the acceptability of the design, construction and maintenance regimes for the prestressing.

4.3.7.3 Description

225 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.

226 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.

227 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 prestressing tendons.
- Bosses and strengtheners around the transfer tube sleeve and equipment hatch.

228 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.

229 Each prestressing 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.

230 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.

231 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.

232 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.

233 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.

234 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 prestressing or when heated, due to geometrical manufacturing defects, remains within acceptable limits.

235 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.

236 The preliminary containment analysis was performed in accordance with ETC-C. The design of the containment has been undertaken by Coyne et Bellier for EDF and AREVA.

4.3.7.4 Findings

237 The containment structure is a pre-stressed, post tensioned structure with the tendons permanently grouted in place following tensioning with a cementicious grout. We 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.

238 These observations were transformed into RO-UKEPR-017 during Step 3.

239 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.

240 Report ENGSGC080361 (Ref. 28) was provided as an initial response to RO-UKEPR-017. The report was reviewed and a meeting held with EDF and AREVA on 2nd April 2009 (Ref. 29). This highlighted a large number of areas where further information was required. A further discussion with EDF and AREVA on 22nd April in Lyon as part of the ETC-C meeting (Ref. 30) settled on those areas where further information was required

241 EDF and AREVA Letter EPR00135R and assorted attachments was provided on 10th July 2009 (Ref. 31) This second response to RO-UKEPR-017 has a much improved coverage of information and greater depth of supporting evidence.

242 A meeting was held on 19th August 2009 with EDF and AREVA to discuss the response, and a number of observations were made. These were passed on to EDF and AREVA (Ref. 32).

243 In order to provide some regulatory clarity to the Issue, I have outlined the basis of a claims argument evidence arrangement which I would expect to see in support of the RO-UKEPR-017 response.

244 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.

245 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.

246 In addition, a suitably detailed ALARP review is required.

247 There are two key areas where we remain to be convinced that the current design can be shown to be acceptable. The primary concern is over the ability to install the prestressing and grouting to a sufficiently high standard that we can have confidence that degradation of the tendons over the lifetime of the structure will not occur. The second is the ability of the monitoring system to identify levels of degradation which threaten the ability of the containment to perform its function when called upon.

248 Grouting of prestressing ducts across a number of industries has been undertaken for over 50 years, with varying levels of success. Concerns over poor quality of grouting led to a temporary ban on this type of construction for highway bridges in the UK for a number of years in the mid 1980s.

249 Once installed, there is little that can be done in terms of inspection (other than semi-destructive) to confirm absolutely that the grout has been installed as intended. There is therefore a strong emphasis on proving through trials that the operations can be undertaken successfully, and that there are suitably robust monitoring arrangements during construction to ensure that due process is followed. The nature of grout means that there are a large number of variables which can influence performance, and it is therefore important to ensure that appropriate controls are in place to minimise deviation from the optimal. In our review thus far, it is unclear what the approach to be taken by EDF and AREVA to this aspect will be. Indeed, the acceptance criteria for the grouting trials has yet to be defined in a suitable manner. Furthermore, there appears to be continual reference to outdated codes of practice for this type of operation.

250 The containment is fitted with a large number of instruments to continually monitor its behaviour. The key measurements for containment integrity are the vibrating wire strain gauges which give a direct measure of the strain condition of the concrete. These instruments are placed during construction and have been demonstrated to be robust (both in France and the UK), and capable of performing successfully for over 40 years. We have no concerns over the nature of the instruments used, however the capability of the system as a whole to perform is unproven. The rationale for the placement of the instruments appears to be historical. We have not seen a concise exposition of what levels of degradation the instrumentation is capable of detecting. In addition, we have not seen a clear description of the sensitivity of the containment as a structure to degradation of the prestressing. Without these two aspects of the design being clear, we cannot come to a meaningful conclusion over the monitoring arrangements.

251 In addition to the EDF and AREVA supplied information, we have participated in witnessing a full scale grouting trial undertaken by VSL in France on a mock up of part of an EPR containment. This was useful in demonstrating the difficulties that can be encountered in performing grouting operations on large deviation tendons and is informing our view of the proposed arrangements for such activities on a UK EPR. Whilst the detailed study over the implementation of a design might be considered beyond the remit of GDA in this case it is fully justified. The premise of the long term integrity of the pressing system is heavily dependant on the quality of the installation, and it is important to gain a high confidence in this aspect of the design at this stage.

252 It is difficult to be definite over the likelihood of EDF and AREVA producing sufficient evidence that we will find the design approach acceptable. It is clear that as yet, we have not been provided with all the arguments and evidence that are either available or capable of being made available. The issue of the acceptability of grouted in place tendons remains outstanding into Step 4.

4.3.7.5 Summary

253 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 judgement to be made over the acceptability of their proposals. These are the grouting operations and the monitoring system for the containment.

254 The issue is being dealt with actively under the terms of RO-UKEPR-037, and will continue into Step 4

255 The Step 4 assessment will focus on the detailed design approach and the use of the finite element codes Precont, Ferrail, ANSYS and SYSTUS as well as the application of the ETC-C to containment design.

4.3.8 Aircraft Protection Structures

4.3.8.1 Scope

256 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.

257 The 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.

4.3.8.2 Standards

258 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.

259 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. 35 will also be considered.

4.3.8.3 Findings

260 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 9/11 event the design was verified and modified as necessary to address the possibility of the direct impact of a large commercial airliner

261 The analysis of the structure has been undertaken using the code EUROPLEXUS, which is discussed in Section 4.3.5. Design guidance is additionally provided in ETC-C.

262 At this stage, limited progress has been made, and a considerable effort will be required within Step 4.

4.3.8.4 Summary

263 Limited progress has been made in this area due to difficulties in the exchange of protectively marked information. This issue has now been resolved. The assessment of the aircraft protection shell will be undertaken as part of Step 4.

4.3.9 Ancillary Structures

264 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.

265 Nonetheless, the classification and overall claims made on the structures can be reviewed at this stage, even if the detailed design cannot be examined.

4.3.9.1 Scope

266 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.

4.3.9.2 Standards

267 There are no particular standards which apply to this review, rather it is a statement of what we have been able to undertake thus far and what we may be able to undertake during Step 4.

4.3.9.3 Findings

268 The CW pumphouse has been designed in detail for the Flamanville and Olkiluoto sites. The intake arrangements for these sites 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. 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 appears to be appropriate, however no further assessment will be undertaken during Step 4.

269 The diesel houses have been designed in detail for the Flamanville and Olkiluoto sites. Their safety classification appears to be reasonable.

270 The detailed design of the service trenches will be a site specific activity, and no further consideration will be given to them.

271 The waste treatment building for the UK application has not been designed as yet. If a design were to emerge during Step 4, then some level of assessment would be undertaken, commensurate with the safety significance of the structure. Otherwise, the assessment of the waste treatment building will be a site specific activity.

272 The Turbine Hall is a not a safety classified structure, however it is seismic class SC2, but has no aircraft protection. This is seen as appropriate, given its role, and that of the equipment contained within it. The design of the structure will be examined in more detail in Step 4. This will focus on the seismic capability of the structure and the arguments around the effects an aircraft impact on the turbine hall and the ensuing secondary damage potential.

273 The design of other ancillary structures which have a safety role has not been undertaken at this stage as the structures will be examined at a site specific level.

4.3.9.4 Summary

274 The bulk of the structures which do not form the Nuclear island will not be examined in detail as part of GDA.

275 A more detailed examination of the Turbine Hall and the Fuel Building will be undertaken in Step 4.

4.3.10 Qualification of Equipment against External Hazards

4.3.10.1 Scope

276 There are a large number of plant items for which qualification against external hazards is required. This will have been done through a mixture of analysis, testing, or experience data. This testing work may have been undertaken over a long period of time.

277 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.

4.3.10.2 Standards

278 The key SAPs (and associated guidance where relevant) 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."

4.3.10.3 Findings

279 Qualification of equipment is discussed in Sub-Chapters 3.1 and 3.6 of the PCSR. Sub-Chapter 3.1 Section 1.2.5.7 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."*

280 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."*

281 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."*

282 The principles outlined above appear to be reasonable. The detailed application of the approach into the design will be undertaken as part of Step 4.

283 Ref. 38 is quoted as the primary document describing the methods for qualification of plant and equipment. This then refers to other IEC and French national standards documents for more detailed aspects. This is seen as a reasonable framework, however there are a number of options which can be chosen as part of the process, and further scrutiny of this will be undertaken in Step 4.

284 A brief review of a small sample of system design manuals showed that there appeared to be a clear definition of the testing requirements. These will be sampled further in Step 4.

4.3.10.4 Summary

285 The general principles for equipment qualification outlined in the PCSR appear to be reasonable.

286 A more detailed investigation into the practical application of the principles will be undertaken as part of Step 4.

4.3.11 PSA Modelling

287 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.

288 The key Steps in the assessment process are as follows:

- 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.

289 An initial review of the PSA is reported elsewhere, and as part of Step 3 only a cursory examination of the documentation has been undertaken.

290 The initial review suggests that probability values for the LOOP from wind are rather lower than expected, as are the probabilities of a loss of the ultimate heat sink from fouling. In addition, the screening approach to external hazards will require further scrutiny in Step 4.

291 The seismic margins assessment in PCSR Chapter 15.6 has been examined at a high level, and would appear to a reasonable approach in principle. A review of the fragility values for key structures and components however has revealed some values which appear to rather higher than anticipated. A more detailed review of the supporting documentation in Step 4 will be undertaken.

4.3.12 Maintenance Inspection and Test

292 The operational lifetime of the EPR is stated as 60 years, with a likely decommissioning period of 40 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.

293 The most important of the tests is the decennial pressure test of the containment, although of equal importance are the integrated leak rate tests and the penetrations undertaken on a rolling basis at refuelling outages etc.

294 Section 3 of ETC-C outlines the overall programme of testing for the containment. It is unclear at this stage what the acceptance criteria are for the containment other than the requirement for elastic reversible behaviour. This is not considered sufficiently well defined and gives too much leeway for acceptance of behaviour which may be indicative of unacceptable degradation. This issue will be addressed as part of RO-UKEPR-017, and be examined in more detail as part of Step 4.

4.3.13 Feedback from Flamanville3 and Olkiluoto 3

295 The construction experience from Flamanville and Olkiluoto provides valuable insight into the buildability of the EPR. They key findings from the projects thus far can be summarised as follows;

- Implementation of complex structural forms is challenging
- On-site activities need clear control structures and authorization routes

4.3.13.1 Flamanville 3

296 The French regulator ASN undertook a site inspection at Flamanville in December 2007. this highlighted a number of concerns including

- Lack of control over concrete heat of hydration tests
- Lack of control over water cement ratios
- Lack of control over the concrete cube samples collected.
- Lack 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.

297 A second inspection by ASN occurred on 8th February 2008. This highlighted the following issues;

- Lack of qualification for the workshop welding the liner plates
- Lack of completeness of the welding log for the liner
- Poor control of crack injections on the base slab
- Unexpected water ingress to the prestressing gallery

These findings were transmitted to EDF by ASN in Letter Ref: Dep-CAEN-0117-2008.

298 A third inspection by ASN occurred on 5th March 2008. This highlighted the following issues

- Lack of control over steel reinforcement placement, inspection and treatment of non-conformances
- Inconsistency between the work plan and the implementation
- Lack of positional checking for inserts following concreting

These findings were transmitted to EDF by ASN in Letter DEP- CAEN-No.0185-2008

299 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.

300 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.

4.3.13.2 Olkiluoto 3

301 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. 33. 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.

4.3.13.3 Implications for GDA

302 The lessons learnt from the FL3 and OL3 construction process can be summarised as follows;

- Strong quality control procedures on site are essential to ensure delivery
- The structures are complex and require careful detailing and strong interaction between the designer and the constructor.

303 Whilst the above may not be seen to be relevant for the GDA process, a more careful consideration shows that there are potential interactions on the following areas.

- Reliability of the design.
- CDM regulations.
- Exclusions from GDA.

4.4 Regulatory Observations

4.4.1.1 Overview

304 Two ROs have been generated in the area of Civil Engineering and External hazards as listed below;

- RO-UKEPR-017 Use of Grouted in Place Prestressing.
- RO-UKEPR-037 Reliability of the ETC-C code.

305 Both of these ROs are a significant challenge to the design philosophy as they either challenge the fundamental standard used for design of the civil structures or the basic philosophy for the integrity of the inner containment.

306 RO-UKEPR-017 is discussed in some detail in Section 4.3.7. The responses to the observation thus far whilst improving our knowledge of the design in some areas have revealed areas where further justification is required. These are principally in the

understanding of the sensitivity of the structure to degradation and how effectively the monitoring regime is at picking this up, and how a high degree of confidence over the quality of installation of the tendons and grouting can be achieved. EDF and AREVA are anticipating presenting this further work before the end of 2009.

307 The implications of failing to satisfy us that the design intentions can be realised is a wholesale re-design of the inner containment, including implications for the Nuclear island foundation raft.

308 RO-UKEPR-037 has been under discussion since April 2009, and after much discussion a set of actions and responses has been agreed. The first stage responses are due by the end of 2009 and the second by the end of March 2010. We have commissioned some additional work by TSCs to provide a more generic consideration of reliability than is being undertaken by the EDF and AREVA response to the RO.

4.4.1.2 Step 4 Strategy

309 It is clear that the response to these 2 ROs will not be available until well into Step 4, and the strategy must incorporate review of the responses in a timely manner.

4.4.1.3 Summary

310 The issues raised in RO-UKEPR-017 and RO-UKEPR-037 are clear and well understood by EDF and AREVA. Clear plans to provide us with responses are in place, and we will need to respond in a timely manner once these responses are received.

4.5 Technical Queries

311 As of the end of September 111 technical queries have been raised in the topic areas of civil engineering and external hazards. Over half of these have been related to ETC-C. These have fallen into the following categories;

- Clarifications of intent/ English
- Request for references
- Request for translation of supporting references.
- Request for background information on the development of ideas and concepts

312 The responses have been generally adequate, although in a number of cases supplementary TQs have been raised. The timeframe to deliver translation of documents, is something which has hindered progress. As a result, we have adopted a subtle shift in approach. This has been to request a wider set of documents in the original French and to then review them at a high level and be more selective in our request for documents or sections thereof.

4.6 Use of Technical Support Contractors

313 The strategy for the Step 3 and 4 assessment has always relied on the use of TSCs to provide assistance. To date, 9 contracts have been placed with 4 organisations. These contracts are providing support in the following areas

- Finite Element Code Assessment
- ETC-C Review
- Load Schedule Review

- Grouted in Place Tendon Review

314 The contractors have actively participated in meetings with EDF and AREVA and arrangements are delivering useful supporting documentation to assist regulatory decision making.

315 Progress of the work has been hampered by the somewhat staggered nature of the delivery of information from EDF and AREVA, primarily a result of the need to translate documents. Work activities are now moving ahead at a reasonable pace, although it is clear that we will need to have delivery of documentation arranged in a more coordinated manner to allow Step 4 to be completed in a timely manner.

4.7 Use of Overseas Regulatory Activity

316 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 (Autorité de Sûreté Nucléaire) and STUK (Säteilyturvakeskus).

317 In addition, the Nuclear Regulatory Commission (NRC) in the United States are currently reviewing the US EPR against their requirements.

318 Thus far, we have undertaken a review of publicly available documents from ASN and STUK. These are discussed further in the following paragraphs.

319 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 No.s, 87,88,89,90 and 91. These are also known as DSR Reports 18,34,69,92 and 103.

320 Thus far only DSR report No. 69 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 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

321 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. 39). This document whilst high level in its content has some useful comments to make on the containment integrity and on the aircraft protection shell.

322 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.

323 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.

324 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.

4.8 Other Health and Safety Regulations

4.8.1 The Construction (Design & Management) Regulations 2007

325 During Step 2, it was noted that there was no reference within the documentation available to the Construction (Design & 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.

326 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

327 In the case of EDF and AREVA, Regulation 12 also apples as below.

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.

328 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.

329 In order to progress this issue, a meeting was held with EDF and AREVA on 20th August (Ref. 34). The outcomes from this meeting are as follows. The designers identified to date are EDF and AREVA. The final design of the power station will be based upon the current plant under construction in France. At the moment EDF and AREVA are ensuring that they are gearing up to meet the requirements of Regulation 11 and Regulation 12 of the CDM with AMEC embedded in the design team to ensure that they are familiar with the obligations under British health and safety law. The initial suggestions are that the plant is being built and designed to the French interpretation of the CDM Regulations.

330 This will be examined in more detail as part of Step 4.

4.9 Interface with EDF AND AREVA

331 During the Step3 assessment a series of meeting have been held with EDF AND AREVA as outlined in the Table below.

Table 3: Summary of Meetings held with EDF and AREVA during Step 3

Meeting No.	Date	Location	Topic
3	25th March 2009	Paris	FE Codes Overview
4	2nd April 2009	Paris	Grouted in Place Tendons
5	21st-22nd April 2009	Lyon	ETC-C
6	11th June 2009	Paris	Code_ASTER
7	23rd June 2009	London	Load Schedule Reliability FE Codes
8	23rd July 2009	Paris	Prestressing Design (Coyne et Bellier)
9	19th August 2009	Lyon	Prestressing Design (Response to RO-UKEPR-017)

332 During each of the meetings, actions are recorded and agreed and tracked through the EDF and AREVA system, which we have full visibility of.

333 As can be seen, regular dialogue has been held with EDF and AREVA. In addition, monthly teleconference discussions on progress of ROs, TQs and meeting actions have been held.

334 In addition, a workshop on the issue of reliability was held with assorted technical support contractors on 2nd September 2009 and EDF and AREVA attended as observers.

5 CONCLUSIONS AND RECOMMENDATIONS

335 The Step 3 assessment of the UKEPR has not progressed as far as initially planned, due mainly to delays in the provision of information and extended timeframes for translation of key documents.

336 The Step 3 approach has been to examine the principles of the design approach to the key safety related structures, with the application to be examined in Step 4.

337 The focus within Step 3 has been on the following areas

- Design Basis Derivation and Load schedule.
- Design Codes.
- Analysis Codes.
- Containment Design.
- Nuclear Island Design.

338 There is a somewhat convoluted justification for the design basis events, partly a result of historical precedents, regulatory requirements and a combination of normal and event based scenarios. The use of multi level hypothesis documents has led to modifications at a low level to the design intent. Despite the non standard approach, I am satisfied that the final design basis events are broadly acceptable. A more structure specific sampling exercise will be undertaken in Step 4.

339 The ETC-C design code has been found to be somewhat lacking at a number of levels. At a concept level, details of its intentions, limitations, inclusion and exclusions are not clear. This has in part led to the development of RO-UKEPR-037. At a lower level, there are a number of poor references, loose statements, poor rigour in terms of design guidance and a general view that the code has not been developed with the same level of scrutiny as would be expected of a nuclear design code.

340 There are a number of detailed technical issues which are emerging from the assessment, primarily linked to the use of modified Eurocodes and / or French national annexe approaches. These will be pursued in Step 4.

341 The analysis codes used form a somewhat complex arrangement, with final design information being the result of influences from up to 6 different codes. The individual codes from the evidence examined thus far appear to be generally suitable for the purposes to which they have been put. Step 4 will examine their interfaces and application in more detail.

342 The containment design utilizing bonded tendons is proving to be a complex and multi-legged assessment task. After some initially unpromising interactions with EDF and AREVA, progress is now being made in providing a more coherent safety justification. There is still a large amount of work to do to convince me that the design is compliant with our regulatory expectations, however achieving this aim is not seen as impracticable.

343 In summary, the assessment is very much work in progress and there is a considerable volume of work to be undertaken in Step 4.

344 It is recommended that the UKEPR design progresses into Step 4 of the GDA.

6 REFERENCES

1 *UK EPR Pre-Construction Safety Report.* UK EPR-0002-132 Issue 02 EDF and AREVA June 2009.

2 *New Reactor Build. Step 3 and 4 Civil and External Hazards Assessment Strategy. ND Division 6* Assessment Report No. AR 08/27, HSE-ND, July 2008. TRIM Ref. 2008/253146.

3 *New Build. Step 2 EPR Civil Engineering and External Hazard Assessment*. ND Division 6 Assessment Report No. AR08/002, HSE-ND, January 2008. TRIM Ref. 2008/40647.

4 *Safety assessment principles for nuclear facilities.* 2006 Edition, Revision 1, HSE December 2006.

5 *Nuclear Power Station Generic Design Assessment – Guidance to Requesting Parties.* Version 3, HSE, August 2008.

6 *Nuclear Divison. Divison 6 Unit 6E Operating Plan 1 April 2008 – 31 March 2009.* TRIM Ref. 2008/215106.

7 *Reactor Safety Reference Levels,* Issue O, Western European Nuclear Regulators Association (WENRA), January 2008.

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

9 Not used.

10 *ND BMS, Technical Assessment Guide – External Hazards.* T/AST/013, Issue 3, HSE, April 2009.

11 *IAEA Safety Standards Series - External Events Excluding Earthquakes in the Design of Nuclear Power Plants.* Safety Guide No. NS-G-1.5, International Atomic Agency (IAEA) Vienna 2003.

12 *IAEA Safety Standards Series - Seismic Design and Qualification for Nuclear Power Plants Safety Guide Safety.* Safety Guide No. NS-G-1.6, International Atomic Agency (IAEA) Vienna 2003.

13 *IAEA Safety Standards Series - Site Evaluation for Nuclear Installations – Safety Requirements.* No. NS-R-3, International Atomic Agency (IAEA) Vienna 2003.

14 *Step 3 EPR External Hazard Assessment Stage 1 Report Overview of External Hazard Loading Claims and their Derivation Methodology.* ABS Report 2120812-R-01, Issue 1, July 2009. TRIM Ref. 2009/231546.

15 *European Utility Requirements for LWR Nuclear Power Plants.* http://www.europeanutilityrequirements.org

16 *Presentation of Approach for Incorporating Hazards in the EPR Project.* Design Report ENSN040070, Revision A1, EDF, October 2008.

17 *ETC-C (EPR Technical Code for Civil Works)* Part 1. EDF, April 2006.

18 *Progress on ETC-C.* HSE Nuclear Directorate Letter EPR70103R, 31st July 2009. TRIM Ref. 2009/297255.

19 http://www.nafems.org

20 *EDF response to meeting action 3.1, verification and validation.* TRIM Ref. 2009/220694.

21 *Meeting with EDF on 2nd April 2009 to discuss grouted in place tendons used in the EPR Design.* ND Division 6 Contact Report No. CR 09/076, April 2009. TRIM Ref. 2009/188341.

22 *Meeting with EDF on 11th June 2009 to discuss FE Codes used in the EPR Design.* ND Division 6 Contact Report No. CR 09/105, June 2009. TRIM Ref. 2009/245316.

23 *Nuclear Island -Overall 3d Model General Description - Nodes, Elements, And Thicknesses.* EDF Report EPR FA3 YR1221 28B01-0102, Revision C, EDF March 2007.

24 *Sizewell B Power Station Station Safety Report,* SXB-IP-772001-718, Section IR5.3(2) Issue 101.

25 *Meeting with EDF and Coyne et Bellier on 23rd July 2009 to discuss PRECONT and FERRAIL programmes used in the EPR Design.* ND Division 6 Contact Report No. CR 09/128, July 2009. TRIM Ref. 2009/295421.

26 *Letter report from Gifford following a visit to Coyne et Bellier to discuss Precont and Ferrail.* TRIM Ref. 2009/380974.

27 *Grouting In Place Prestressing Tendons Deliverable D1, Review Of Relevant Literature, Causes Of Degradation, Best Practice And Monitoring.* Gifford Report No. 15194/G01, June 2009. TRIM Ref. 2009/220664.

28 *Prestressing System for UK EPR inner containment.* Note ENGSGC080361, EDF, March 2009.

29 As Ref. 21.

30 *Meeting with EDF on 21st-22nd April 2009 to discuss ETC-C Code used in the EPR Design.* ND Division 6 Contact Report No. CR 09/075, April 2009. TRIM Ref. 2009/186420.

31 *Further Response to Civil Structures RO-UKEPR-17 on Grouted Tendons.* EDF Letter ND(NII), EPR00135R, August 2009. TRIM Ref. 2009/276877.

32 *Comments sent to EDF following the meeting on 19th August.* TRIM Ref. 2009/347437.

33 *Management of Safety Requirements In Subcontracting During The Olkiluoto 3 Nuclear Power Plant Construction Phase*, STUK Investigation Report 1/06.

34 *L4 Site Preparation Meeting.* ND Division 6 Contact Report No. EDF Licensing CR 09/005. 20th August 2009. TRIM Ref. 2009/332576.

35 *Engineering Safety Aspects of the Protection of Nuclear Power Plants Against Sabotage.* IAEA Nuclear Security Series No. 4, Feb 2007. International Atomic Energy Agency (IAEA), Vienna, 2007.

36 *EPR – External Hazards – Inventory of Combined Events with Internal Faults and/or other (Internal and External) Hazards taken into account in Design.* Design Report ENSNEA080058, Revision A, EDF, November 2008.

37 *A standard for standards. Specification for structure, drafting and presentation.* BS 0-3:1997. British Standards Institute, 1997.

38 *Design and Construction Rules for Electrical components of nuclear islands.* AFCEN Publication RCC-E, December 2005.

39 *Safety assessment of the Olkiluoto 3 (OL3) Nuclear power plant Unit for the Issuance of Construction Licence.* STUK Report, 2005. http://www.stuk.fi/ydinturvallisuus/ydinvoimalaitokset/ydinvoimalaitosluvat/viides/en_GB/rakentamislupa/_files/12222632510016763/default/olkiluoto3_rakentamisluvan_turvallisuusarvio.pdf

Table 4

Safety Assessment Principles to be Considered During Step 3

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
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

SAP Number	SAP Title	Assessed Category
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	
RP.6	Shielding	S3
RW -	Storage of Radioactive Waste	
RW.5	Storage and Passive Safety	S3
DC -	Decommissioning	
DC.1	Design and Operation	S3

Note: S2 = Step 2

S3 = Step 3 and 4

Figure 1

Mind Map of Safety Assessment Principles

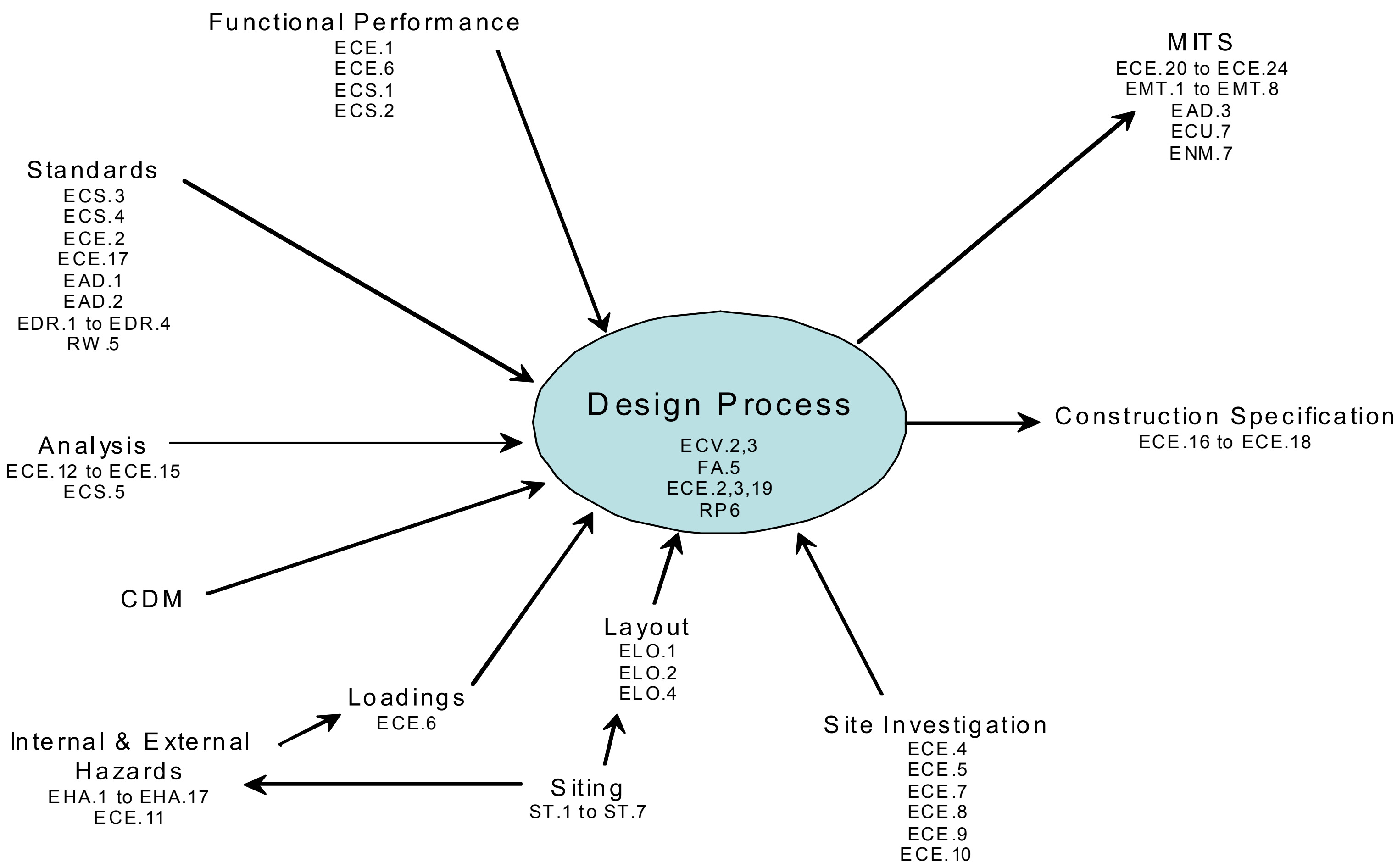


Figure 2

Map Showing Finite Element Code Coverage in the UK EPR Design

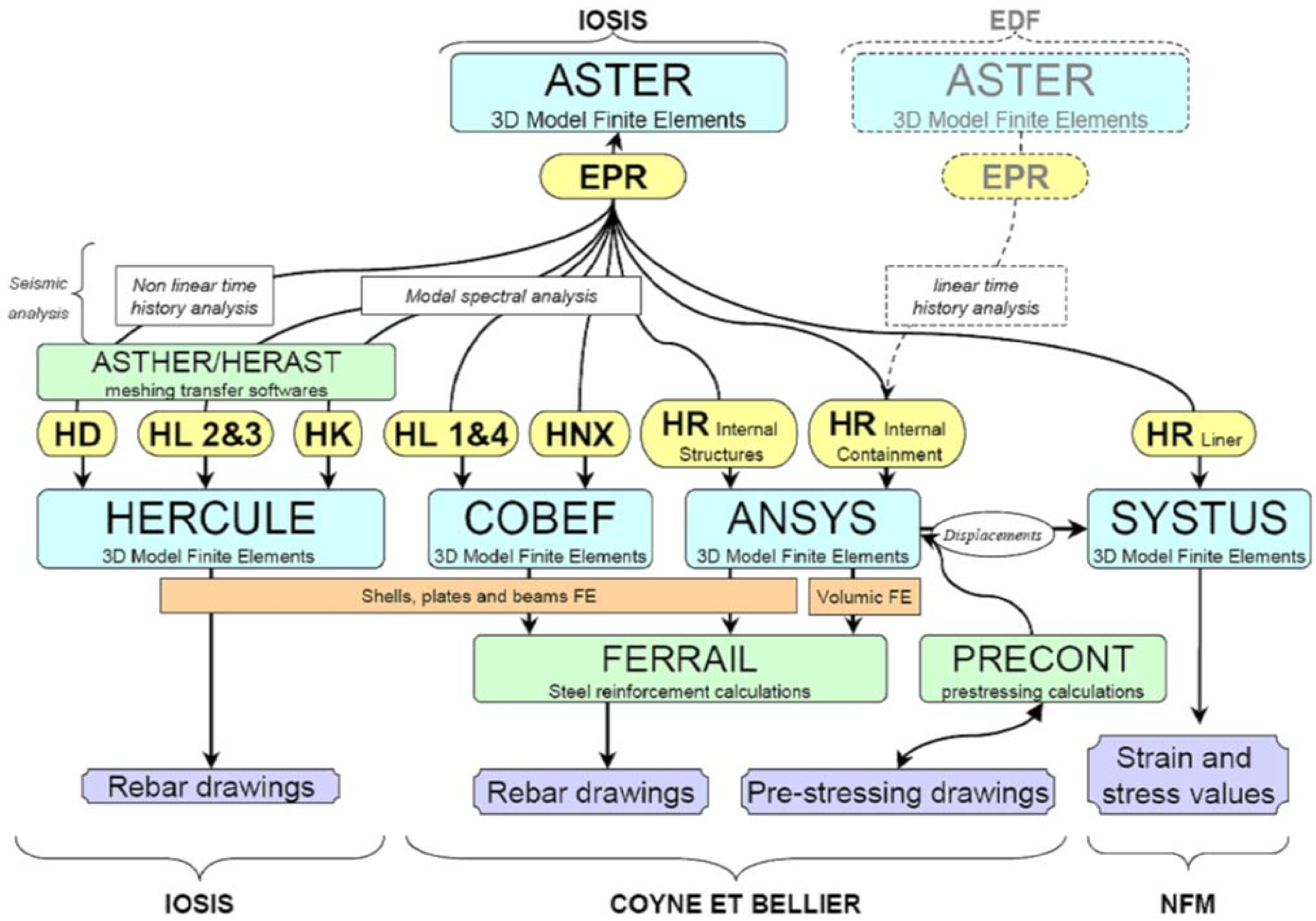
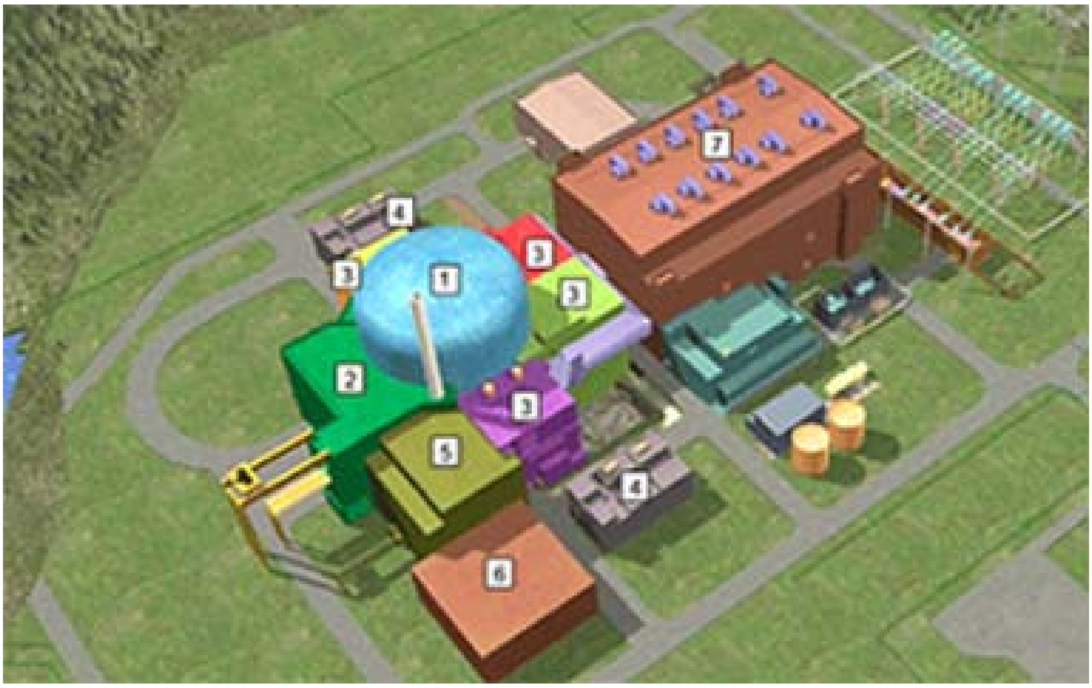


Figure 3

Typical Layout of an EPR



Key

1 Reactor Building

2 Fuel Building

3 Safeguards Buildings

4 Diesel Building

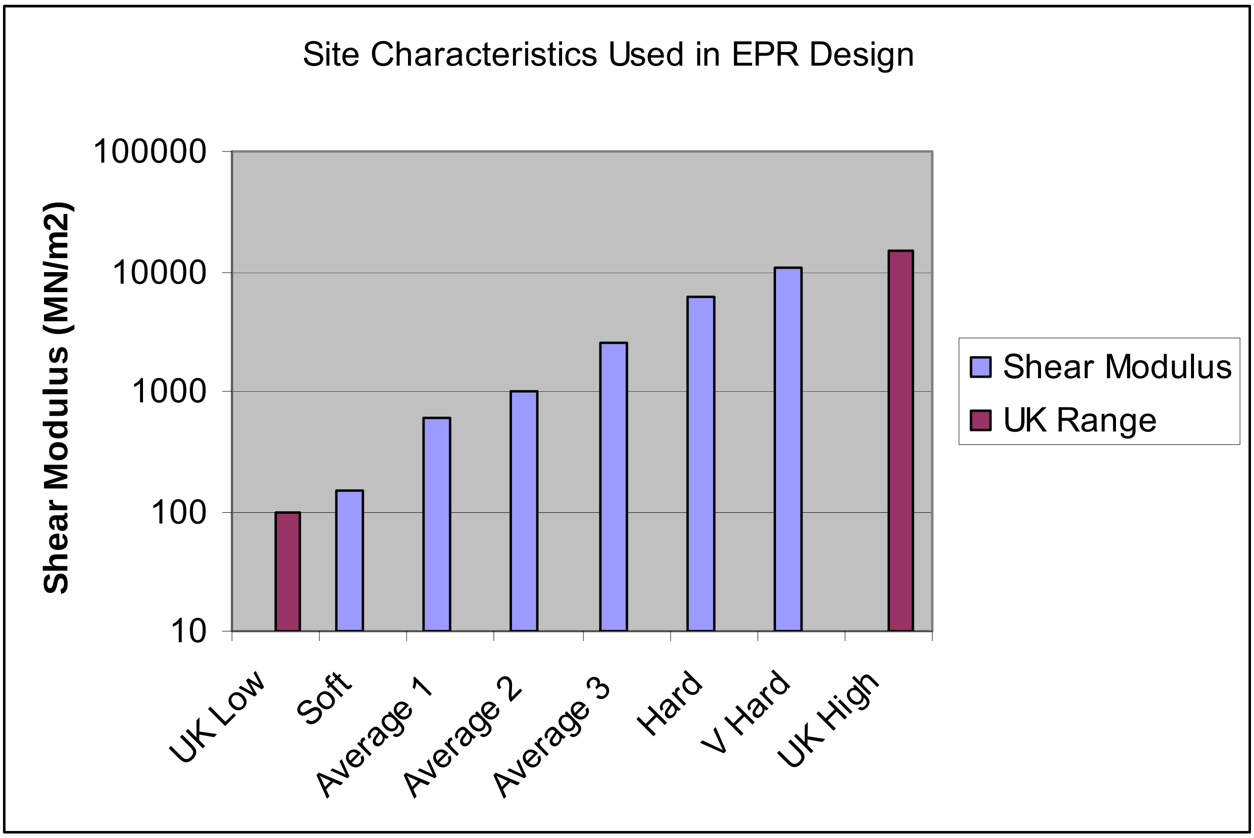
5 Nuclear Auxiliary Building

6 Waste Treatment Building

7 Turbine Hall

Figure 4

Site Soil Characteristics Used in the EPR design and UK Envelope



Level 1 EDF

Level 2 EDF

Level 3 Sofinel

Level 4 (Design offices)

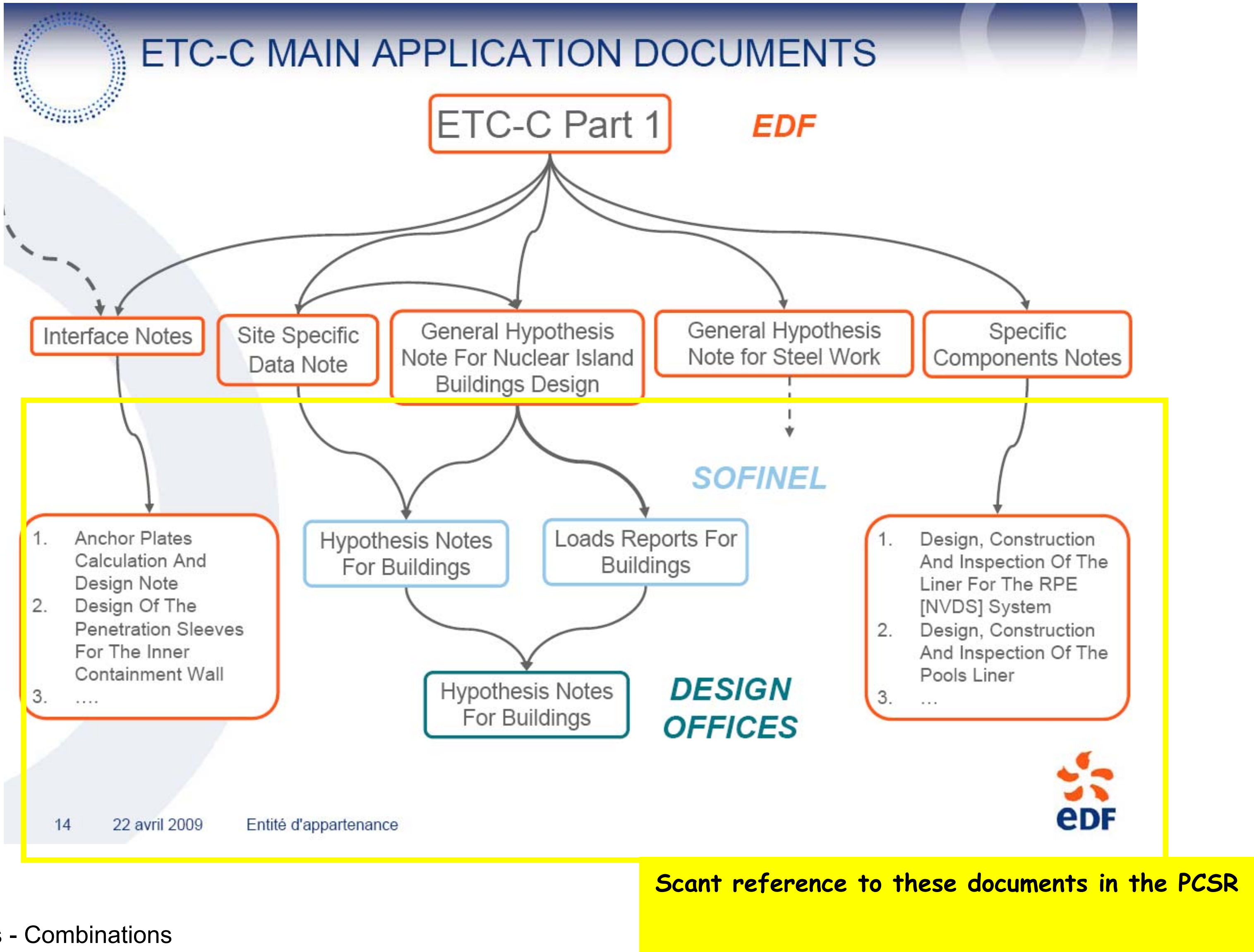


Figure 5: Principal Data Sources - Combinations

Annex 1 – Civil Engineering and External Hazards – Status of Regulatory Issues and Observations

RI / RO Identifier	Date Raised	Title	Status	Required timescale (GDA Step 4 / Phase 2)
RO-UKEPR-017		Grouted in place prestressing tendons	The initial response has been received, and a meeting held on 19th August 2009. Comments have been sent to EDF AND AREVA and a response awaited.	Step 4
RO-UKEPR-037		Reliability of the ETC-C	The actions have been agreed and timescales. No formal response has been received as yet.	