NUCLEAR DIRECTORATE

GENERIC DESIGN ASSESSMENT – NEW CIVIL REACTOR BUILD

STEP 3 MECHANICAL ENGINEERING ASSESSMENT OF THE EDF AND AREVA UK EPR

DIVISION 6 ASSESSMENT REPORT NO. AR 09/014-P

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EXECUTIVE SUMMARY

This report presents the findings of the mechanical engineering assessment of the EDF and AREVA UK EPR Pre-Construction Safety Report (PCSR) undertaken as part of Step 3 of the Health and Safety Executive's (HSE) Generic Design Assessment (GDA) process.

The Step 3 assessment process consists of examining the arguments and identifying the evidence in the EDF and AREVA submission relating to the mechanical engineering aspects, and assessing them against the expectations and requirements of the Safety Assessment Principles (SAPs), legislation, good engineering practice, internal Nuclear Directorate (ND) guidance and relevant information from external bodies, i.e. The Western European Nuclear Regulators' Association (WENRA) reference levels and The International Atomic Energy Agency (IAEA) standards and guidance.

The Step 3 aim for the mechanical engineering assessment is to:

- Review the level of design completeness.
- Assess relevant aspects of the safety case.
- Assess the scope and extent of claims and arguments presented.
- Consider whether the mechanical design aspects are likely to meet regulatory expectations.
- Consider overseas regulators' knowledge of the designs.
- Consider the scope of, and plan for, further assessments.
- Liaise with the Environment Agency to aid their public consultation process.

Mechanical engineering covers a broad range of equipment types, and the assessment approach up to and including Step 3 has been to review selected equipment based on our regulatory expectations in terms of their safety functions. The results of this assessment approach are reported in this Step 3 report. This assessment approach considers, and challenges, the safety function categorisation and equipment classification philosophies adopted by EDF and AREVA, and draws conclusions as appropriate.

The assessment is to review the Structures, Systems and Components (SSCs) for their:

- Safety function categorisation and equipment classification.
- Design and reliability claims.
- Equipment Qualification, to deliver its safety function.
- Capability to satisfy the demands in normal operation and a fault scenario.
- Access and egress to enable: operations, inspections, testing, maintenance and equipment replacement.
- Interfaces with other assessment topic areas.
- Design completeness.

At this stage of the GDA process good progress is being made in terms of reviewing the EDF and AREVA submission, identifying issues and areas for more detailed review and discussion, and progressing these to a satisfactory conclusion.

A number of Technical Queries (TQ) have been raised, and responses received, which have been reviewed as part of the assessment process. Three technical meetings have been held with EDF and AREVA and further direct interactions have taken place via telephone conferences, and also technical presentations.

A degree of confidence has been gained in the Equipment Qualification processes applied by EDF and AREVA, specifically from the description of the Stand Still Seal System as an example.

A degree of confidence has also been gained in the design process applied by EDF and AREVA, based on the assessment undertaken to date. However, the present safety function categorisation and equipment classification methodologies do not at present align with the expectations described in the UK SAPs. Further work is underway in this area by EDF and AREVA, which will attract an appropriate degree of assessment as necessary. The definition of safety functional requirements for mechanical items important to safety, the degree to which this is promulgated from assembly down to component design, and then captured and retained through the design and implementation lifecycle, are also areas of continued regulatory interest.

At this stage of the overall GDA process, no Regulatory Observations (RO) or Regulatory Issues (RI) have been identified associated with the EDF and AREVA submission.

FOREWORD

Mechanical Engineering

In carrying out this assessment, the term 'mechanical engineering' encompasses Structures, Systems and Components (SSCs) that generally contain dynamic elements and interfaces. This is to distinguish it from the discipline of structural integrity, which is concerned with SSCs which are static in nature, primarily focussing on containment pressure boundaries. Not withstanding this definition, a number of static components will also be of interest to the mechanical engineering discipline, and subject to appropriate assessment.

Examples of dynamic components that are considered to be of interest include:

- Control Rod Drive Mechanisms.
- Pumps.
- Valves, (check valves, motor operated valves, safety relief valves and containment isolation valves).
- Cranes.
- Mechanical handling systems.
- Nuclear Ventilation (HVAC).
- Diesel generators.

Examples of static components that are considered to be of interest include:

- Heat exchangers.
- Gloveboxes, cabinets.
- Transport packages.
- Stillages.
- Seals.
- Strainers.
- Component support structures.

Structural integrity aspects with reference to the containment pressure boundaries and containment vessel internals are not specifically considered or assessed under the mechanical engineering discipline. These aspects are the subject of assessment under the discipline of Structural Integrity and reported in the assessment report covering that topic.

LIST OF ABBREVIATIONS

ALARP	As Low As Reasonably Practicable
BMS	(Nuclear Directorate) Business Management System
CCWS	Component Cooling Water System
CRDM	Control Rod Drive Mechanism
CVCS	Chemical and Volume Control System
EA	The Environment Agency
EBS	Extra Boration System
EDF and AREVA	Electricité de France SA and AREVA NP SAS
GDA	Generic Design Assessment
HEPA	High Efficiency Particulate Air
HSE	The Health and Safety Executive
IAEA	The International Atomic Energy Agency
LOCA	Loss of Coolant Accident
MFWS	Main Feed Water System
MSSS	Main Steam Supply System
ND	The (HSE) Nuclear Directorate
NPP	Nuclear Power Plant
PCER	Pre-Construction Environment Report
PCSR	Pre-Construction Safety Report
PDMS	Piping Design Management System
RI	Regulatory Issue
RIA	Regulatory Issue Action
RCCAs	Rod Cluster Control Assemblies
RCS	Reactor Coolant System
RO	Regulatory Observation
ROA	Regulatory Observation Action
RP	Requesting Party
SAP	Safety Assessment Principle
SBO	Station Black Out
SIS/RHRS	Safety Injection System/Residual Heat Removal
SSCs	Structures, Systems and Components
SSSS	Stand Still Seal System
TAG	(Nuclear Directorate) Technical Assessment Guide
TQ	Technical Query
WENRA	The Western European Nuclear Regulators' Association

TABLE OF CONTENTS

1 INTRODUCTION

1 This report presents the findings of the mechanical engineering assessment of the EDF and AREVA UK EPR Pre-Construction Safety Report (PCSR) (Ref. 1) undertaken as part of Step 3 of the HSE Generic Design Assessment (GDA) process. This assessment has been undertaken in line with the requirements of the Business Management System (BMS) document AST/001 (Ref. 2) and its associated guidance document G/AST/001 (Ref. 3). AST/001 sets down the process of assessment within the Nuclear Directorate (ND) and explains the process associated with sampling of safety case documentation. The Safety Assessment Principles (SAPs) (Ref. 4) have been used as the basis for the assessment of the PCSR associated with the EDF and AREVA submission. The SAPs set the regulatory expectation that all credible hazards on a nuclear power plant or nuclear chemical plant site are identified and considered in safety assessments. Ultimately, the purpose of assessment is to reach an independent and informed judgment on the adequacy of a nuclear safety case and associated design.

1.1 Assessment Scope

2 In carrying out this assessment, the term mechanical engineering encompasses Structures, Systems and Components (SSCs) that generally contain dynamic elements and interfaces. This is to distinguish it from the discipline of structural integrity, which is concerned with SSCs which are static in nature, primarily focussed on containment of the relevant pressure boundaries. Not withstanding this definition, a number of static components will also be of interest to the mechanical engineering discipline, and have been assessed as appropriate.

3 Examples of dynamic components that are considered to be of interest include:

- Control Rod Drive Mechanisms.
- Pumps.
- Valves, (check valves, motor operated valves, safety relief valves and containment isolation valves).
- Cranes.
- Mechanical handling systems.
- Nuclear Ventilation (HVAC).
- Diesel generators.

4 Examples of static components that are considered to be of interest include:

- Heat exchangers.
- Gloveboxes, cabinets.
- Transport packages.
- Stillages.
- Seals.
- Strainers.
- Component support structures.

5 Structural integrity aspects with reference to the containment pressure boundaries and containment vessel internals are not specifically considered or assessed under the mechanical engineering discipline. These aspects are the subject of assessment under the discipline of Structural Integrity (Ref. 6).

6 This assessment report formally records the mechanical engineering progress statement in support of the Generic Design Assessment (GDA) Step 3 against the EDF and AREVA EPR Design submission (Ref. 1).

7 It should be noted that the mechanical engineering topic was not specifically assessed during Step 2, and other disciplines did not raise any issues specifically related to the mechanical aspects during their Step 2 assessment work.

8 The objective of Step 3 is to move from examination of the fundamentals in terms of claims made by EDF and AREVA, to assessing the engineering design, principally at the system level. This Step 3 assessment has been guided by analysis of the EDF and AREVA supporting arguments to underpin their claims, and then moving into the identification of supporting evidence contained within the EDF and AREVA submission.

9 The Step 3 assessment process consists of examining the arguments and identifying the evidence in the EDF and AREVA submission relating to the mechanical engineering aspects, and assessing them against the expectations and requirements of the Safety Assessment Principles (SAPs) (Ref. 4), legislation, good engineering practice, internal ND guidance and relevant information from external bodies, i.e. Western European Nuclear Regulators' Association (WENRA) reference levels (Ref. 7) and International Atomic Energy Agency (IAEA) standards and guidance (Ref. 8).

10 The Step 3 aim for the mechanical engineering assessment is to:

- Review the level of design completeness.
- Assess relevant aspects of the safety case.
- Assess the scope and extent of claims and arguments presented.
- Consider whether the mechanical design aspects are likely to meet regulatory expectations.
- Consider overseas regulators' knowledge of the designs.
- Consider the scope of, and plan for, further assessments.
- Liaise with the Environment Agency to aid their public consultation process.

11 The principal deliverables from the mechanical engineering Step 3 assessment are a progress statement and further definition of the assessment scope going forward.

12 Site specific aspects and commissioning are excluded from the assessment during this Phase. This includes any aspects specifically associated with construction of the power station and site-specific operational matters; these aspects are to be considered during Phase 2.

2 NUCLEAR DIRECTORATE'S ASSESSMENT

2.1 EDF and AREVA Safety Case

13 A safety case is generally assessed by identifying the claims on structures, systems and components, and people, and then assessing the associated arguments and underpinning evidence. This assessment structure, which should be aligned to the safety case structure, is essentially a 'top down' approach and provides a logical framework to ensure that all hazards have been adequately identified and suitably addressed.

14 The nature of mechanical engineering, and associated mechanical engineering assessment, favours an alternative 'bottom up' type approach. In this case mechanical items important to safety are identified and then assessed on the basis of their safety function, categorised in functional terms as associated with either cooling, reactivity control, or containment.

15 Mechanical engineering covers a broad range of equipment types and the assessment approach up to and including Step 3, has been to review selected equipment based on our regulatory expectations, in terms of their safety functions. The results of this assessment approach are reported in this Step 3 report. This assessment approach considers, and challenges, the safety function categorisation and equipment classification philosophies adopted by EDF and AREVA, and draws conclusions as appropriate.

16 This assessment approach will further interface with the approach adopted by other disciplines, including coordination with the areas of Fault Studies and Probabilistic Safety Assessment, as well as Human Factors as necessary, to provide a holistic assessment in terms of claims, arguments and evidence covering mechanical engineering items important to safety.

17 Based on the above approach, the following table provides a summary of my determination of the EDF and AREVA safety case in respect of mechanical equipment, which has guided my assessment.

Table 1 - Summary of determination of the EDF and AREVA safety case in respect of mechanical equipment

No.	Primary Safety Function	SSCs	Safety Aspect
1	Reactivity Control	Control Rod Drive Mechanism (CRDM)	The moderator / coolant contains soluble boron as a neutron poison. The boron concentration in the coolant is varied as required to make relatively slow reactivity changes, including compensation for the effects of fuel burn-up. Additional neutron poison (gadolinium), in the form of burnable-poisoned rods, is used to establish the required initial core reactivity and power distribution. The core reactivity and the core power distribution are also controlled by movable Rod Cluster Control Assemblies (RCCAs), which are neutron absorber rods that enable rapid changes in reactivity to be made. Each RCCA consists of a group of individual absorber rods fastened at the top end to a common hub or spider assembly. The RCCAs are split into several groups. The Control Rod Drive Mechanisms (CRDMs) move the RCCAs and enable them to be dropped, to remain as they are, or to be withdrawn.
2	Reactivity Control	Extra Boration System (EBS)	Emergency addition of Boric acid provides a diverse method of shutting the reactor down.
3	Reactivity Control	Chemical and Volume Control System	During normal operation, plant start-up, and plant shutdown conditions the Chemical and Volume Control System (CVCS) must, in conjunction with the Reactor Boron and Water Makeup System, regulate and adjust the Reactor Coolant System boron concentration to control power changes (in conjunction with the control rods) and to offset reactor fuel burn-up.
4	Heat transfer / Residual heat removal	Safety Injection System / Residual Heat Removal	The Safety Injection System / Residual Heat Removal System (SIS/RHRS) is a combined system providing safety injection and removal of residual heat from the reactor.

No.	Primary Safety Function	SSCs	Safety Aspect
		System	The SIS/RHRS consists of four separate, independent trains, each of these trains being able to inject borated water into the primary circuit by means of an accumulator, a medium-pressure safety injection pump (MHSI) and a low-pressure safety injection pump (LHSI) with a heat exchanger at its outlet. The system also provides controlled heat extraction from the primary circuit in shutdown mode, chiefly the residual power in the core, through the LHSI pump and heat exchanger and the heat exchanger's bypass line.
5	Heat transfer / Residual heat removal	Component Cooling Water System	The Component Cooling Water System (CCWS) must contribute to the following main functions: • decay heat removal from the primary system: cooling of SIS pumps and heat exchangers in the reactor normal cooling phase (Residual Heat Removal) or during incident or accident conditions; • decay heat removal from the spent fuel pool; • heat removal from the safety chilled water system refrigeration plants.
6	Heat transfer / Residual heat removal	Plant Gas System	Nitrogen distribution system - this system is used to supply nitrogen to maintain the SIS accumulator tanks under pressure. Nitrogen is also used as part of the Stand Still Seal System (SSSS) for the reactor coolant pump sealing system during shut down.
7	Heat transfer / Residual heat removal	Chemical & Volume Control System	Under certain small break Loss of Coolant Accident (LOCA) conditions, the CVCS helps maintain the required water inventory in the Reactor Coolant System (RCS).
8	Heat transfer / Residual heat removal	Reactor Coolant System	During normal operations the RCS transfers the heat generated in the reactor to the secondary loop system. The reactor coolant pump rotor equipped with its flywheel provides sufficient inertia to ensure the appropriate flow rate, and therefore sufficient Departure from Nucleate Boiling margins before the automatic shutdown of the reactor in the event of a reactor coolant pump coast-down transient condition.
9	Heat transfer / Residual heat removal	Main Feedwater System	When operating at power the Main Feedwater System (MFWS) must contribute, with the main steam system circuit, to removing the heat produced by the reactor core. The MFWS must maintain the level of water in the steam generators at the required value and within limits compatible with the protection systems during steady state and normal operating transients to remove core decay heat.

No.	Primary Safety Function	SSCs	Safety Aspect
10	Heat transfer / Residual heat removal	Main Steam Supply System	In normal operation, the Main Steam Supply System (MSSS) must remove decay heat by transferring steam to the condenser, from power operation to the connection of Residual Heat Removal System. Under certain fault events, the MSSS must remove decay heat by dumping steam into the atmosphere to allow safe shutdown to be reached.
11	Containment of radioactive substances	Main Steam Supply System	The Main Steam Supply System must contain the activity of the primary system in the event of Steam Generator Tube Rupture by isolating the affected Steam Generator on the steam side.
12	Containment of radioactive substances	Containment Isolation	The Combustible Gas Control System contributes to the safety function 'containment of radioactive substances' by ensuring: • Limitation and reduction of loads on containment structures caused by hydrogen combustion. • Reduction and limitation in hydrogen mole fraction during Loss of Coolant Accidents (LOCA), as well as to prevent any risk of combustion in the containment. • Reduction in mean and local hydrogen concentration during severe accidents to ensure containment integrity.
13	Containment of radioactive substances	Ventilation Building Containment	The EPR reactor building consists of a cylindrical reinforced concrete outer shield building, a cylindrical pre-stressed concrete inner containment building with a steel liner, and an annular space between the two buildings. The shield building protects the containment building from external hazards. The reactor shield building functions as a secondary containment to prevent the uncontrolled release of radioactivity to the environment following a postulated design basis accident. The reactor shield building and annulus ventilation system are designed to provide the secondary containment function under the environmental conditions of normal operation, maintenance, testing, and postulated accidents, including protection against the dynamic effects associated with a design basis accident. The annulus is maintained at a sub-atmospheric pressure during normal operations and following postulated design basis accidents, establishing an essentially leak-tight barrier against uncontrolled

No.	Primary Safety Function	SSCs	Safety Aspect
			release of radioactivity to the environment.
		Nuclear Auxiliary Building Ventilation System	The Nuclear Auxiliary Building Ventilation System and its extension, the Fuel Building Ventilation System operate continuously. They are designed for the following purposes: • to keep the ambient conditions within limits prescribed for correct operation of equipment and / or staff in normal operation (air supply and filtering, heating / refrigeration / humidity); • to ensure during normal operation that contamination is contained at source to avoid its spreading from potentially contaminated areas to potentially less contaminated areas; • to reduce the concentration of aerosols and radioactive gases in the atmosphere; • to keep a negative pressure in the Nuclear Auxiliary Building and the Fuel Building compared to the outside pressure using an automatic control damper in the air supply trains.
		Internal Filtering	The Internal Filtering System operates during operation of the plant, in order to reduce the concentration of radioactive iodine and aerosols in the reactor building.
		Ventilation in the controlled area of the safety buildings	The ventilation systems have the following safety functions: •to maintain static and dynamic containment under normal operating conditions and fault scenarios. •to filter extract air (for particulate and iodine) under normal operations and fault scenarios.
		Ventilation in the Main Control Room	The functional role of the ventilation system of the Main Control Room is as follows: • to maintain acceptable ambient conditions (temperature and humidity) for staff and equipment in the Main Control Room; • to ensure habitability of the Main Control Room, the Technical Support Centre and associated rooms, even in the event of radioactive contamination of the environment.
		Diesel Room Ventilation System	For the main diesel rooms and the Safety Black Out diesel rooms, the safety roles of the ventilation system are as follows: • to maintain an ambient temperature below a specified maximum by removal of the heat released during operation of the diesel and of electrical components; • to maintain an ambient temperature above a specified minimum in tank rooms, I&C rooms, battery and electrical rooms.

No.	Primary Safety Function	SSCs	Safety Aspect
		Ventilation of the Controlled Area of the Operating service centre Ventilation of the Controlled Area of the Effluent Treatment Building	The safety function of the ventilation system is to prevent and minimise radioactive releases from the hot laboratories in the Operational Service Centre. The safety function of the ventilation system is to prevent and minimise radioactive releases from the Effluent Treatment Building.
14	Containment of radioactive substances	Component Cooling Water System	Provide a barrier against leakage of fluid from primary containment and reactor systems.
15	Containment of radioactive substances	Reactor Coolant System (RCS)	During normal operations the RCS transfers the heat generated in the reactor to the secondary loop system. The RCS acts as the second containment barrier of defence following the fuel cladding. The Reactor Pressure Vessel (EDF and AREVA) seal arrangement provides a containment barrier. The RCS pump seal provides a containment barrier. The Pressuriser Safety Relief Valves limit the pressure within the RCS to meet the overpressure protection requirements. Spring loaded Safety Relief Valves protect the Residual Heat Removal System during cooldown.

2.2 Standards and Criteria

18 The approach is to carry out this assessment in accordance with :

- ND standards;
- applicable SAPs;
- guidance of the Technical Assessment Guides (TAGs).

19 This approach ensures the assessment provides a targeted, consistent and transparent consideration on the adequacy of the EDF and AREVA design.

20 The mechanical engineering assessment is to be carried out with the aid of a number of applicable SAPs, which are principles against which regulatory judgements are made and provide fundamental guidance in scoping an assessment topic and in carrying out an effective assessment.

21 Generally SAPs capture the requirements of WENRA reference levels and the IAEA Standards Series requirements. If a requirement is not found to be covered by a SAP the assessor will include the requirement within the assessment (Ref. 5).

22 It is worth noting, the nature of the mechanical engineering discipline often drives the assessment down to component level. Assessment at this component level can be extremely wide ranging given the very large number of such components, with numerous interfaces, across various plant process systems and covering several disciplines. As a consequence, a wide range of SAPs and TAGs can be applicable to carrying out an effective assessment. The approach to carrying out an effective sampled assessment is to select the most appropriate SAPs and TAGs to a particular selected mechanical engineering aspect.

23 The assessment of mechanical engineering aspects is guided by this selection of relevant SAPs. In making a judgment on whether a SAP is applicable to a mechanical engineering aspect, consideration is given to the following factors;

- Key Principles.
- Safety Categorisation, Classification and Standards.
- Design and Reliability.
- Maintenance, Inspection and Testing.
- Layout.
- Pressure Systems.
- Integrity of metal Components and Structures.
- Safety Systems.
- Containment and Ventilation.

24 Annex 2 Table A2.1 lists and interprets the SAPs that are considered applicable to carrying out an effective mechanical engineering assessment.

25 Annex 2 Table A2.2 lists the TAGs that are considered applicable to carrying out an effective mechanical engineering assessment.

2.3 Assessment Methodology

26 The assessment methodology for executing the assessment was to carry out the assessment in accordance with the Project Implementation Document (Ref. 5).

27 The Assessment was carried out on a sampling basis, dictated by consideration of risk and hazard significance, in coordination with the other assessment disciplines and early mechanical engineering assessment findings. The assessment has focused on the primary safety functions identified from within the EDF and AREVA submission, as described in Table 1. The GDA sampling policy requires the whole design to be considered, and then assessment targeted on specific areas based on considerations of their hazard and risk.

28 The initial assessment was to briefly assess the 'Fundamental Safety Overview' and the claims being made within the EDF and AREVA submission.

29 With resource and programme constraints, the assessment policy focus on the primary safety functions that manage the:

- Reactivity control.
- Heat transfer and removal.
- Containment of radioactive substances.

30 The progress statement has been prepared from:

- Reading the appropriate chapters of the EDF and AREVA PCSR submission.
- Holding the appropriate technical discussions with interfacing disciplines.
- Consideration of international acceptable standards.
- Consideration of operational data and findings.
- Consideration of other regulators' findings.
- Raising and issuing of Technical Queries, followed by assessment of EDF and AREVA responses.
- Holding the necessary technical meetings to progress the identified lines of enquiry.

31 The assessment considered the Structures Systems and Components (SSCs) for their:

- Design completeness.
- Safety categorisation and classification.
- Design and reliability claims.
- Equipment Qualification and integrity to deliver their functionality.
- Capability to satisfy their safety functions in normal operations and in fault scenarios.
- Layouts, access, ingress and egress to enable: operations, inspections, testing, maintenance and equipment replacement to be carried out.
- Interfaces with other assessment topic areas.

32 The assessment was carried out in accordance with the ND standards against the applicable SAPs and with the guidance of the Technical Assessment Guides (TAGs).

33 The GDA of the EDF and AREVA submission has been undertaken across 15 key topic areas. As part of the coordination of the assessment process, discussion with the technical leads in each of the key areas has been undertaken as necessary.

34 The GDA has reviewed the overall safety of the design. The PSA has been undertaken to identify the reliability claims on each SSC and the Deterministic Safety Analysis (Fault Studies) has been undertaken to identify the equipment performance required by the safety case.

2.4 Design Status

35 As part of my assessment it was necessary to understand the design status of mechanical SSCs that are important to safety, to enable an effective mechanical engineering assessment to be scheduled and carried out.

36 I identified the listed SSCs to be of a regulatory interest, due to their correlation with the primary safety functions (Table 1).

- Control Rod Drive System (CRDM).
- Reactor Coolant System (RCS).
- Compressed Air System.
- Nuclear Island HVAC System.
- Cranes and Handling Systems.
- Transport Packages.

- Building Layouts, provision of ingress and egress routes for the replacement of mechanical items that are important to safety.

37 My assessment has found:

- The concept design for the above SSCs is sufficiently developed to allow my Step 3 assessment to be carried out.
- A significant quantity of information supporting my mechanical engineering assessment is captured within the project System Design Manuals (SDMs).
- The EDF and AREVA design process is to complete the design in parallel to the building of a plant. With the GDA assessment information typically being based on the Flamanville Nuclear Power Plant (NPP) the availability of design information diminishes as you progress away from the nucleus of the plant.
- Noting the above strategy, for several items, functional and qualification tests are outstanding, for example:
  - i) Extra Boration System (EBS) pumps.
  - ii) Reactor Coolant Pump (RCP).
  - iii) Pressure relief valves.
  - iv) Valves dedicated to severe accidents.
  - v) HVAC items important to safety.
  - vi) Polar crane aspects that are important to safety.
- Although a number of functional and qualification tests are outstanding against mechanical engineering components, I consider this is likely to have a minimal impact on carrying out an effective GDA for mechanical engineering. However these may require to be considered in Phase 2.
- During my Step 3 assessment the Reactor Coolant Pump Stand Still Seal System Equipment Qualification process was specifically assessed, with the findings recorded under the Reactor Coolant System Pump section within this report.

38 My assessment of mechanical engineering aspects associated with specific types of equipment is generally recorded against the individual SSCs, further in this report.

2.5 Design Process

39 EDF and AREVA need to demonstrate that mechanical items important to safety follow a robust design process. A robust design process provides the evidence and the auditable trail that these items will achieve their design intent.

40 My assessment process has involved reading the EDF and AREVA submission, the issue of Technical Queries and undertaking technical meetings to inform the progress statement.

41 My assessment considers:

- The techniques and tools utilised to ensure the safety requirements are clearly identified, categorised, classified, cascaded and substantiated throughout the project life cycle with an adequate audit trail.
- That good engineering practice is captured in the mechanical design from the generation of the concept and through the project life cycle.
- The management of plant and equipment layout and interfaces.

42 As specific examples, I identified the:

- Valve selection process as an area for more detailed assessment. Mechanical valves have important safety functions in NPPs to manage reactivity control, heat transfer and removal, and containment of radioactive substances. I therefore considered it appropriate to target my assessment in this area.
- The Reactor Coolant Pump (RCP) as an area for more detailed assessment. The RCP has important safety functions in NPPs regarding its ability to ensure adequate heat transfer. I therefore considered it appropriate to target my assessment in this area.

This equipment is discussed in greater detail later in this report.

2.5.1 Safety Categorisation and Classification

43 During the EPR basic design phase, five main safety functions were considered in developing the functional classification:

- Control of fuel integrity at power and core reactivity.
- Control of Reactor Coolant System water inventory.
- Control of Reactor Coolant System temperature.
- Control of Reactor Coolant System and Steam Generator pressure.
- Control of the confinement of radioactive materials.

44 Design evolution changed this to three main safety functions:

- Control of reactivity and control of the containment.
- Control of heat removal and control of the containment.
- Control of the containment.

45 An example of UK safety function categorisation is:

- Category ‘A’ any function that plays a principal role in ensuring nuclear safety.
- Category ‘B’ any function that makes a significant contribution to nuclear safety.
- Category ‘C’ any other safety function.

46 An example of UK SSC safety classification is:

- Class 1 any structure, system or component that forms a principal means of fulfilling a Category ‘A’ safety function.
- Class 2 any structure, system or component that makes a significant contribution to fulfilling a Category ‘A’ safety function, or forms a principal means of ensuring a Category ‘B’ safety function.
- Class 3 any other structure, system or component.

47 The EDF and AREVA PCSR is not explicit in linking the main safety functions with the EPR functional classification levels.

48 At the first technical meeting, EDF and AREVA presented their design process that substantiates the item of equipment design criteria and safety classification. The presentation focused on the RCP and the Medium Head Safety Injection Pump.

49 I noted the ‘M’ classification relates to the pressure boundary requirements, ‘SC’ relates to the seismic aspects and other functional safety aspects are captured under the ‘F’ classification.

50 The classification is typically against the whole assembly and not broken down to a detailed component level. EDF and AREVA indicated that this aspect is captured in a separate document.

51 I considered that clarification is required regarding the understanding of the verification process for safety categorisation allocation.

52 At the 2nd technical meeting EDF and AREVA identified that the Operating Technical Specification is a key document, which identifies all safety functional requirements for the facility, plus also identifies Operating Rules and Operating Instructions in accordance with safety and regulatory requirements. This document is currently unavailable, as it is being developed. I will consider reviewing the process for developing Operating Rules and Instructions and available documentation, during my further assessment.

53 With mechanical engineering typically focusing on individual items/components within a system, there is a requirement to understand the safety functional requirements at a reasonably detailed level. Therefore the absence of this information is currently limiting the depth of assessment.

54 At this stage the assessment is progressing, gaining a degree of confidence in the EDF and AREVA methodology in the allocation of safety categorisation and classification to items that are important to safety. However, it should also be noted at this stage that discussions are ongoing with EDF and AREVA in respect of the Safety Function Categorisation, and equipment Classification philosophy across all disciplines, which at present does not match the expectations as described in the UK SAPs. I understand that EDF and AREVA has recognised the need to review their philosophy and present their arrangements in a format more aligned to UK regulatory expectations, which can then be reviewed across all assessment disciplines. I will review progress in this area and draw conclusions as appropriate.

55 I consider there is a requirement for EDF and AREVA to provide:

- Further clarification regarding categorisation and classification methodologies, with examples of SSC classification and safety function categorisation.
- Further evidence of adequacy of ‘M’ classifications (relating to containment) to ensure safety functions are achieved.
- An updated PCSR to capture the above.

56 Assessment of specific mechanical items’ safety functional requirements are captured under individual mechanical items assessment areas, reported later in this document.

57 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

2.5.2 Transfer of Safety Requirements through the Project Life Cycle

58 At the 1st. mechanical engineering technical meeting EDF and AREVA presented an overview of the System Design Manuals and their role through the different phases of the project life cycle.

59 The availability of English translated System Design Manuals (SDM) limited assessment during the early part of the Step 3 process. Although the majority of documents are now translated, it is worth noting that future requests for translations need to be done promptly, otherwise the translation task alone may limit the depth of assessment that can be carried out under the GDA.

60 The SDMs evolve as the project progresses through its life, through to the operational stage. Stage 1 captures the concept requirements, Stage 2 captures the detailed design

aspects with Stage 3 capturing the 'As Built' status following commissioning. I am satisfied with:

- The stated methodology for evolving SDMs as the project progresses through the project life cycle.
- The content captured under Stage 1, which is expected to be of an adequate level to carryout the GDA.

61 EDF and AREVA retain Design Authority status throughout the different phases of the project life.

62 Safety functions are typically specified at an assembly level and are not broken down to the detailed component level. There is a requirement to further understand how the safety functional requirements are being cascaded through the project life cycle.

63 The presentation provided an initial level of confidence in the principle for transferring requirements to the supply chain.

64 Additional assessment findings are captured under individual component assessment areas.

2.5.3 Good Engineering Practice

65 Due to my prioritisation, this assessment topic is at a relatively early stage and will be further progressed, with consideration to the:

- Evidence that the design process encompasses operational experience.
- Arguments and design criteria for incorporating flexible connections (as a sampled aspect).

66 Additional assessment findings of good engineering practice will also be captured under individual component assessment areas.

67 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

2.5.4 Layout / Interfaces

68 The layout of mechanical plant and equipment can affect the safety of plant. I have considered the adequacy of ingress and egress provision for carrying out inspection, maintenance, replacement and testing of mechanical items that are important to safety.

69 Due to my prioritisation, this assessment topic is at a relatively early stage and will be further progressed, with consideration to the assessment of the replacement sequence of an RCS pump (as a sampled area). This is due to its size, mass and location within the plant.

70 At the 2nd technical meeting EDF and AREVA presented an overview of their Piping Design Management System (PDMS) model, which is used to support their design process. The model, although not verified, is utilised to develop the NPP design, to understand interfaces and space management aspects.

71 EDF and AREVA explained that the PDMS database model feeds into the NavisWorks visualisation tool, and the model covers the entire facility, and not just the nuclear island. EDF and AREVA stated that the present model is for the Flamanville project, (which is the approach adopted for the GDA process), but changes as necessary will be incorporated for the UK facilities. They also explained that the PDMS model is used to generate some design drawings.

72 In response to ND questioning, it was noted that mechanical equipment identified within the model contains some engineering attributes but insufficient to allow direct procurement. However the model can generate piping isometrics and cable tray drawings directly from the model.

73 EDF and AREVA explained that PDMS is primarily a space management tool, it is not used to perform calculations etc, which are undertaken using separate processes.

74 A discussion took place on the provisions within the design to enable mechanical components to be replaced and for the model to demonstrate the sequence of activities. I highlighted that this was an area of mechanical engineering regulatory interest.

75 EDF and AREVA stated that a number of large components are installed through temporary construction access apertures and if these components require to be replaced during operations they would be subjected to a stand-alone study at that time to determine the methodology.

76 The demonstration of a filter change involved the use of 'cat ladders' and the temporary assembly of lifting equipment that overhung hand railing. This is not normal practice on a UK licensed site and may be an area of future regulatory interest.

77 The presentation demonstrated a 3D model that is a useful aid in support of developing NPP design concepts, understanding interfaces and space management aspects.

78 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

2.6 Specific Structures, Systems and Components

79 Based on the stated assessment methodology, assessment is being carried out on a sampling basis, dictated by consideration of risk and hazard significance. Table 1 identifies the SSCs that I consider support the primary safety functions of:

- Reactivity control.
- Heat transfer and removal.
- Containment of radioactive substances.

80 The following Structures, Systems and Components have therefore been identified for specific mechanical engineering assessment during Step 3, and this is reported as follows.

2.6.1 Control Rod Drive Mechanisms

81 The Control Rod Drive Mechanisms (CRDMs) have an important safety function of controlling the core reactivity.

82 Against the background that CRDMs are of an established principle of design and with significant operational experience within NPPs around the world, my assessment philosophy is to focus on the:

- Safety design improvements, associated claims, arguments and evidence.
- Safety categorisation and classification.

83 My assessment considered the CRDM latch assembly as being a particular item important to safety and should therefore be categorised and classified accordingly. Initial assessment of the safety documentation has not substantiated this aspect to my satisfaction.

84 The latch unit is located within the lower part of the pressure housing. It is the actual component, which converts the magnetic forces generated by the coils, located outside the pressure housing into sequences of mechanical motion. In principle, it consists of three armatures which alternatively engage two groups of latches into the grooves of the drive rod, thus holding the RCCA in position or moving it up or down to manage reactivity control.

85 At the 1st mechanical engineering technical meeting EDF and AREVA presented an overview of the response to the TQ associated with the CRDM design.

86 I noted the EPR CRDM design has evolved out of the Konvoi CRDM design, which has of the order of 35 years of operational experience within the German NPPs.

87 Several aspects of the CRDMs have been the subject of review, with the aim of improving the CRDM design and understanding the design life limits.

88 Examples of such aspects include:

- Increased seismic values.
- Introduction of a displacement limiter.
- Revised design of the Fuel to CRDM coupling.

89 A CRDM test trial is in progress, and uses representative process parameters. The trial has shown the CRDM to complete 7.5 million cycle steps. The trial is continuing until the CRDM fails.

90 Assessment to date has provided confidence that:

- the CRDMs are of an established design;
- reliability is underpinned from historical operational data and from carrying out the research trials;
- design improvements are typically of a minor nature.

91 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

2.6.2 Valves

92 Several valves have important safety roles and functions in managing:

- Reactivity control.
- Heat transfer and removal.
- Containment of radioactive substances.

The types of valves supporting these roles include:

- containment, isolation valves;
- check valves;
- motor operated isolation valves;
- safety relief valves.

2.6.2.1 Valve Selection Process

93 I selected the Reactor Cooling System as a system that contains several valves, of various types, which are important to safety.

94 At the 1st. mechanical engineering technical meeting EDF and AREVA presented an overview of a selection diagram that is utilised for the selection of a particular valve type.

95 I noted the existence of a design procedure that aids the valve selection process and forms part of the audit trial.

96 There is minimal evidence that the valve selection process is the subject of a review and a valve is typically selected based on the allocated project engineer's judgement and experience.

97 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

2.6.2.2 Containment, Isolation Valves

98 My assessment has identified the role of valves associated with the Reactor Coolant System (RCS) as being important to achieving containment and isolation.

99 One Technical Query (TQ) has questioned the explanation of the operation of the pilot valves used in the design of the Main Steam Isolation Valve.

100 The response to this TQ has been reviewed as part of the assessment process, and the following conclusion has been drawn appropriate to this stage of the overall GDA process:

- I am satisfied at this stage with the response received in respect of the apparent anomaly regarding the energisation logic for the pilot valves associated with the Main Steam Isolation Valve on the secondary side. This has confirmed that the diagram notes in Chapter 10.3 of the PCSR, (UKEPR-0002-103 Issue 01) were in error, and I am satisfied that these notes have now been corrected in the re-issue of the PCSR, (UKEPR-0002-103 Issue 02).

101 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

2.6.2.3 Check Valves

102 My assessment identified the use of check valves for process containment isolation on the Emergency Boronation System.

103 At the 1st mechanical engineering technical meeting EDF and AREVA presented an overview of the response to the TQ associated with the check valve design.

104 I noted that valve selection was on the basis of:

- Utilisation of passive items of equipment within containment.
- Equipment diversification to perform containment isolation.

105 EDF and AREVA stated the design justification was from operational experience and the carrying out intrusive inspection and testing during scheduled shutdowns.

106 At the 2nd mechanical engineering technical meeting EDF and AREVA presented an overview of the procedure that demonstrates the functionality of a safety related check valve. The examples presented were the Extra Boronation System and Safety Injection System check valves.

107 I noted the PCSR Chapter 18 is scheduled for an update. EDF and AREVA explained that the update shall capture the subject of periodic testing of check valves that are important to safety.

108 EDF and AREVA presented the procedure that is followed to verify that a check valve is functional for both flow and seating.

109 EDF and AREVA stated that periodicities for valve maintenance were identified through a process of expert judgement, accounting for Operational Experience Feedback, plus reliability studies and requirements as necessary.

110 EDF and AREVA stated the check valve under discussion is not disassembled as part of the periodic test. This is contrary to my understanding from the 1$^{st.}$ technical meeting and I may consider further assessment in this area.

111 I considered the presentation provided positive evidence that the check valves can achieve their safety function design intent.

112 Assessment to date has only seen safety functional requirements identified at an assembly level. I have not seen evidence of safety functional requirements specified at a detailed component level.

113 Operational experience of Sizewell "B" NPP has indicated that non return valves are reliable at achieving their design criteria. When the passing of a valve is a potential issue, the design typically incorporates two non return valves in series. Experience has only found components being subject to excessive wear when the process flow fluctuates, which causes the valve seat to flap excessively throughout the process cycle.

114 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

2.6.2.4 Safety Relief Valves

115 The safety relief valves used within the primary circuit and the secondary circuit are important areas for regulatory attention, as part of the Step 3 assessment. In particular, the claims and arguments in respect of these safety relief valves are important in regard to the frequency of spurious opening, the reliability of operation on demand (expressed as a probability of failure on demand), and the reliability of re-seating following operation.

116 As part of the sampling process undertaken (which is intrinsic to the assessment process), I identified the following safety relief valves for initial consideration:

- the Pressuriser Safety Relief Valve design used on the primary circuit;
- the safety relief valve design used in the main steam line on the secondary side;
- the spring loaded safety relief valve design used within the Residual Heat Removal System.

117 Technical Queries were raised in early July 2009 to seek arguments and evidence in relation to the recent operational experience of these valve types. These queries also sought background information regarding some changes made to the detailed design of the Pressuriser Safety Relief Valve, compared to those used on a large number of German power stations, plus changes to the secondary side steam relief system, compared to the N4 and Konvoi design configurations.

118 The responses to these Technical Queries have been reviewed as part of the assessment process. I am satisfied at this stage with the responses received in respect of the specific design of:

- Pressuriser safety relief valve utilised in the primary circuit. I note that a new spring loaded pilot is proposed for the EPR in order to improve leak tightness.

- Safety relief valve used in the main steam line on the secondary side. EDF and AREVA have stated that no occurrences of spurious opening, refusal to open, or failure to re-seat have been reported with the design proposed for the EPR.
- Safety relief valve used within the Residual Heat Removal System. The valve design proposed for the EPR is spring loaded, but fitted with hydraulic dampers. The feedback from operating French plants with similar damped valves has identified no reported instances of spurious opening, refusal to open, or failure to re-seat.

119 Further responses will be reviewed as appropriate in due course.

120 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

2.6.3 Reactor Coolant System Pump

121 The role of the Reactor Coolant Pump (RCP) is important to managing the primary safety functions of :

- Reactivity control.
- Heat transfer and removal.
- Containment of radioactive substances.

122 My assessment has taken into consideration responses to Technical Queries and discussions from technical meetings. Further assessment will also be informed by information relating to operational experience and the carrying out of a maintenance inspection of the RCP seal replacement at Sizewell “B” NPP.

123 At the 1st mechanical engineering technical meeting EDF and AREVA presented an overview of the response to the TQ associated with the Stand Still Seal System (SSSS) design.

124 I noted the introduction of the SSSS is a design improvement. The SSSS provides a static containment seal. This addresses the potential issue of common cause failure of both seal injection from the Chemical Volume Control System (CVCS) and the thermal barrier on the loss of electric power.

125 The reactor coolant pump SSSS prevents excessive loss of coolant along the shaft in the event of failure of the normal seals and hence contributes to the RCS containment safety function.

126 The SSSS automatically closes when the reactor coolant pump shutdown is detected by the pump rotational speed measurement sensors, in combination with simultaneous loss of seal cooling from the CVCS and the CCWS.

127 The SSSS is hydraulically actuated via a nitrogen seal gas supply, and disabled via a seal gas relief to the Containment Building atmosphere.

128 All seal leak-off lines from the reactor coolant pumps are automatically isolated by closure of the motorised isolation valves on No. 1, 2 and 3 seal leakage recovery lines.

129 The SSSS is classified as a F1B system and hence the single failure criterion can be considered at the level of its function. The corresponding functional redundancy is provided by the thermal barrier cooling by the CCWS, also classified F1B.

130 The shaft sealing system comprises three seals arranged in series and a SSSS.

131 The design of seals N° 1 and N° 2 is identical to that used on the N4 and CP 1300 plant reactor coolant pump assemblies, which have had good operational experience. The design of seal N° 3 is very similar to that used on 900 MW plants’ reactor coolant pumps.

132 Some improvements have however been adopted to comply with the EPR specification:

- Pump operation at low pressure when the System Injection System, Residual Heat Removal System is connected to the RCS and is operating in residual heat removal mode.
- Inclusion of a SSSS in the absence of a back-up system for the rapid injection at the shaft seals in the event of total loss of electrical power.
- The SSSS, which can be activated when the pump is shutdown.

133 Consequently, the shaft sealing system and the SSSS are fitted with 'O' rings manufactured with a grade of material qualified for high pressures and temperatures.

134 Seal N° 1 makes the major contribution to the pressure drop with a controlled leak-off, discharged to the CVCS. Under normal operation, this seal is fed from the CVCS, via the injection line at seal N° 1. In the event of loss of water injection from the CVCS, it is fed by reactor coolant cooled by the thermal barrier heat exchanger.

135 Seals N° 2 and 3 provide the remaining pressure drop, with a negligible leak-off discharged to the vent and drain system.

136 Seal N° 2 acts as a back up to seal N° 1 in the event that the latter fails, and is designed to provide seal function for at least thirty minutes with the pump rotating and for 24 hours when the pump is shut down. A failure in seal N° 1 is detected by the flow meter located on the leak-off line. In this event the leak-off line is automatically isolated, and the reactor power is reduced to an acceptable level, allowing the faulty reactor coolant pump to be shut down. Once the reactor coolant pump is shutdown, the plant is then shutdown, the SSSS is activated and all other leak-off lines are closed.

137 The SSSS is located on the upper section of the shaft sealing system, above seal N° 3.

138 Once the pump is shut down, a piston ring, actuated by a low-pressure nitrogen supply, closes the air gap between the shaft and this ring, and creates a leak-tight metal to metal surface contact. This ensures that the shaft is leak-tight once the pump is shut down and all the leak-off lines closed (these lines are closed off in the following order: seal N° 3, seal N° 2 and lastly seal N° 1).

139 The SSSS is designed to be leak-tight in the event of:

- Simultaneous loss of water supply from the CVCS and the CCWS used to cool the shaft sealing system during a Station Black Out (SBO).
- A cascade failure of all stages in the shaft sealing system.

140 In the event of a SBO, the SSSS and the leak off line isolating valve for the three shaft seals are automatically closed once the reactor coolant pump is shutdown.

141 The standstill seal system is designed to:

- Close, once activated, in the event of the pressure and temperature resulting from SBO.
- Isolate a significant leak, which would result from a cascade failure of the shaft seals.
- Prevent damage to the shaft sealing system in the event of inadvertent closure of the standstill seal when the pump is running.
- Prevent auto-closure in the event of a cascade failure of the shaft seal sealing system.
- Remain leak-tight once activated until a very low reactor coolant pressure is reached, even if the nitrogen supply pressure is lost.

142 At the 2nd. mechanical engineering technical meeting EDF and AREVA presented an overview of the Stand Still Seal System, covering the design functionality, design status and the adopted design qualification process.

143 The EDF and AREVA presentation identified that the incorporation of the SSSS came from operational experience of the Konvoi Nuclear Power Plant. EDF and AREVA also explained that this design is incorporated into all EPR designs, i.e. Flamanville, OL3 etc.

144 The presentation described the design process methodology adopted to evolve and substantiate the design, the development trials carried out, and that due consideration had been given to the operational process parameters.

145 In response to questioning, EDF and AREVA stated:

- There is no impact on safety if the SSSS was inadvertently activated, although there could be an operational impact.
- The principle of operation of the SSSS for the EPR was the same as for the German reactors, but the detailed design was different, which had driven the requirement for further qualification testing.
- The qualification tests had been undertaken using irradiated seals as appropriate. It was also stated that seals are inspected at every outage, and are replaced at a nominal periodicity of 5~6 years.

146 I noted the qualification tests had been undertaken for both hot (thermal) and cold conditions, and had covered the 24 hour test period as necessary to qualify the station Black Out specification.

147 EDF and AREVA explained that the SSSS is only activated in the event of a fault, and had no duty function; (and as such was a Safety System in UK terminology).

148 My assessment has considered the seal pump principle against an alternative canned pump principle. Although the canned pump has the advantage of eliminating the potential Loss of Containment Accident (LOCA) at the seal face, the seal principle is well understood for such an application with the design evolving with many years of application in NPPs around the world. Other aspects such as component inspection and maintenance are judged as less demanding, with the majority of the components being located outside the primary pressure boundary.

149 To date I have only seen safety functional requirements identified at an assembly level. I may consider reviewing the evidence of how safety functional requirements are achieved at a more detailed component level.

150 Assessment to date has provided confidence that the RCP is following a satisfactorily design process.

151 I consider from a Regulatory point of view, a seal pump is an acceptable principle that has extensive historical evidence of satisfactory operation data on various NPPs across the world.

152 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

2.6.4 Cranes

153 The cranes utilised throughout the proposed nuclear facility are important items for regulatory attention. As part of the Step 3 assessment process a number of faults are worthy of consideration in respect of cranes which are used for nuclear use, which can challenge the safety functions of cooling, criticality control, and containment. For cranes which are located inside buildings, typical faults are associated with the load path (with

the potential to lead to dropped or suspended loads), including double blocking, snagged loads, ledged loads, rope failures, gearbox and motor failures, failures associated with the braking systems and failures associated with the control and protection systems. A common feature for cranes required to undertake nuclear lifts, is that they are 'single failure proof', such that no single failure will result in loss of capability of the system to retain the load. Dual rope systems are also commonly employed as part of achieving this criterion, with energy absorbing systems incorporated into the designs as necessary. Furthermore, cranes are commonly de-rated against their industrial code capacity as a specific safeguard to minimise the probability of failure.

154 As part of the Step 3 assessment, I have asked a number of Technical Queries relating to cranes identified within the facility as having a significant nuclear use. Specifically queries have been asked in relation to the Refuelling Machine and the Polar Crane.

155 The responses to these Technical Queries have been reviewed as part of the assessment process. I am satisfied at this stage with the responses received in respect of the:

- Dynamic loads predicted in the remaining rope in the event of failure of one rope, in dual rope systems.
- Energy absorbing systems for the polar crane.
- General description of the control and protection systems for the re-fuelling crane and the polar crane.
- Design and protection systems to account for faults associated with the re-fuelling machine and the polar crane.
- Braking systems associated with the re-fuelling machine and the polar crane.

156 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

2.6.5 Nuclear Ventilation (HVAC)

157 Nuclear ventilation systems have an important nuclear safety function in terms of supporting containment of nuclear material, by ensuring that air movements and discharges are adequately directed and filtered to reduce doses to operators and the public under both normal and accident conditions. Nuclear ventilation systems commonly use HEPA (High Efficiency Particulate Air) filters on the discharge line to capture airborne particulate containing radioactivity, as a means of minimising discharges to meet statutory requirements including the ALARP principle. Ventilation also plays an important role in ensuring the habitability of the nuclear facility under normal and accident conditions, with a specific focus on the habitability of the Main Control Room under accident conditions. The principles of nuclear ventilation are well understood, and as a matter of principle, for dynamic containment, there should be a cascade of air flow from areas of lower to those of higher potential contamination, to control the spread of contamination throughout the facility, and to support the correct segregation of areas from a worker dose perspective.

158 As part of the Step 3 assessment process, I have raised Technical Queries associated with the following issues of nuclear safety significance:

- The design and type of HEPA filters which are used within the design, noting that difficulties are often encountered with achieving an adequate seal within the filter housing. In addition, modern HEPA filters commonly use a safe change filter change system, using an integral bag system, to provide containment during the maintenance activity.

- The habitation of the Main Control Room.
- The design criteria for the various ventilation systems, relating to the reasonably foreseeable rise in temperature over the ~ 60 year period to account for global warming.
- The durability of external features of the ventilation systems, accounting for the UK maritime climate likely to be experienced at the proposed UK reactor sites.

159 The responses to these Technical Queries have been reviewed as part of the assessment process. I am satisfied at this stage with the responses received in respect of the:

- Design, testing and maintenance arrangements associated with the HVAC system HEPA filters. I note that HEPA filters use bag system change arrangements as necessary.
- Maximum air temperature rating for the HVAC system, noting the responses to previous TQs which have also addressed this area. I note the response indicates that the detailed identification of facility design temperatures will be undertaken as part of the acquisition of site data as part of the site engineering activities and of the Nuclear Site Licence application. I also note the response from EDF and AREVA that in the event of higher than design temperatures being experienced, then no cliff edge effects are anticipated, although some loss of system performance may be experienced. I consider this response to be reasonable.
- Durability of the HVAC external features. Specifically EDF and AREVA has described the use of stainless steel for HVAC features for external and adjacent equipment, to minimise any corrosion effects.

160 EDF and AREVA also provided a technical presentation to the ND/EA regulatory assessment team in the UK on the 17th July 2009, which helped to provide a greater understanding of the HVAC design configurations and principles, and which provided confidence in their design process.

161 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

2.6.6 Gloveboxes / Cabinets

162 Gloveboxes and mechanical equipment cabinets are an area for limited regulatory interest as part of the assessment process. Interest in this area primarily relates to protection of the operator, although there is the potential that further derivative issues may arise through progression of the assessment activity.

163 As part of the Step 3 assessment activity, EDF and AREVA has been asked to identify the gloveboxes and cabinets within the facility design, identifying their safety functional requirements and associated ventilation systems, plus the standards used for their design and fabrication.

164 These responses will be reviewed as appropriate in due course.

165 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

2.6.7 Heat Exchangers

166 Heat exchangers used within the facility have an important safety function in terms of cooling, and I have identified the following heat exchangers for assessment at this stage:

- Residual Heat Removal System Heat Exchangers.
- Spent Fuel Pool Heat Exchangers.

167 Specifically I have raised technical queries to clarify the arguments and evidence in relation to the recent operational experience of the specific design of heat exchangers proposed within the design, which relates to the reliability of operation of these engineering features.

168 The response to these queries has stated that the principles of operation and the method of design and manufacture of this equipment are conventional and similar to those applied in previous nuclear projects in France and Germany. Furthermore, the response states that the Residual Heat Removal System Heat Exchangers and Spent Fuel Pool Heat Exchangers of the N4 Nuclear Power Plants, (and more generally those of the French fleet), have the same overall design, and that no negative operational feedback has been recorded so far.

169 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

2.6.8 Diesel Generators

170 Diesel generators are traditionally designated as part of a safety system. They typically provide a diverse means of providing AC power to support the operation of components that are important to safety. They are accordingly assigned with the appropriate safety categorisation and classification.

171 From my initial assessment line of enquiry, I have determined that EDF and AREVA has assigned a safety claim on the diesel generators.

172 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

2.6.9 Spent Fuel Handling, Pond Stillages, Radioactive Waste Containers and Transportation Flasks

173 I have identified that the following mechanical items:

- Spent fuel handling equipment.
- Pond Stillages.
- Radioactive Waste containers.
- Transportation flasks.

are important in supporting the primary safety functions of cooling, criticality control and containment, and are therefore areas of Regulatory interest.

174 However, my assessment to date has focused on the structures, systems and components that are directly associated with the main reactor island primary safety functions. This has resulted in limited progress being made in this assessment area.

175 I note the EDF and AREVA design process may limit the availability of information associated with some aspects of the mechanical items in this area. This EDF and AREVA adopted design strategy may limit the ability to carry out a significant depth of assessment under the GDA process due to the available level of information.

2.6.10 Component Cooling Water System (CCWS)

176 Auxiliary systems that support components of a system important to safety should be considered part of that system and should be classified accordingly unless failure does not prejudice successful delivery of the safety function.

177 The Component Cooling Water System provides a supporting function to several mechanical items that are important to safety and is therefore an area of regulatory interest. The CCWS also has a fundamental safety function in its own right to provide cooling to the primary circuit as part of the Residual Heat Removal System.

178 My line of enquiry is to understand the mechanical components that are important to safety, which have a reliance on the CCWS.

179 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified in this area.

3 CONCLUSIONS

180 At this stage of the GDA process good progress is being made in terms of reviewing the EDF and AREVA submission, identifying issues and areas for more detailed review and discussion, and progressing these to a satisfactory conclusion.

181 A number of Technical Queries have been raised, and responses received, which have been reviewed as part of the assessment process. Three technical meetings have been held with EDF and AREVA and further direct interactions have taken place via telephone conferences, and also technical presentations.

182 A degree of confidence has been gained in the Equipment Qualification processes applied by EDF and AREVA, specifically from the description of the Stand Still Seal System as an example.

183 A degree of confidence has also been gained in the design process applied by EDF and AREVA, based on the assessment undertaken to date. However, the present safety function categorisation and equipment classification methodologies do not at present align with the expectations described in the UK SAPs. Further work is underway in this area by EDF and AREVA, which will attract an appropriate degree of assessment as necessary. The definition of safety functional requirements for mechanical items important to safety, the degree to which this is promulgated from assembly down to component design, and then captured and retained through the design and implementation lifecycle, are also areas of continued regulatory interest.

184 At this stage of the overall GDA process, no Regulatory Observations or Regulatory Issues have been identified associated with the EDF and AREVA submission.

4 REFERENCES

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

2 *ND BMS, Assessment Process*. AST/001, Issue 2, HSE, February 2003.

3 *ND BMS, Guide: Assessment Process*. G/AST/001, Issue 2, HSE, February 2003.

4 *Safety Assessment Principles for Nuclear Facilities*. 2006 Edition, Revision 1, HSE, January 2008.

5 *New Reactor Build. Generic Design Assessment. Step 3 Mechanical Engineering.* ND Division 6 Assessment Report No. AR08/034 (Project Implementation Document) HSE December 2008. TRIM Ref. 2009/21999.

6 *Generic Design Assessment – New Civil Reactor Build. Step 3 Structural Integrity Assessment of the EDF and AREVA UK EPR.* ND Division 6 Assessment Report No. AR09/012 HSE November 2009. TRIM Ref. 2009/305940.

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

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

Annex 1 – Mechanical Engineering – Status of Regulatory Issues and Observations

RI / RO Identifier	Date Raised	Title	Status	Required timescale (GDA Step 4 / Phase 2)
Regulatory Issues				
None				
Regulatory Observations				
None				

Annex 2 – Mechanical Engineering Tables – Applicable Safety Assessment Principles and Technical Assessment Guides

Table A2.1 lists and interprets the SAPs that are considered applicable to carrying out an effective mechanical engineering assessment. Noting mechanical engineering covers a wide range of components, not all SAPs are applicable to the assessment of each individual component. The policy is to select the applicable SAPs for the component that is being assessed.

The third column in Table A2.1 cross-references to the associated Technical Assessment Guide. The fourth column highlights the Step during which the assessment is initiated (Phase 1, Step 3). The fifth column highlights the associated reference to the WENRA reference levels and the sixth column highlights the associated reference to the IAEA safety standard series requirements (Ref. 8).

Table A2.1 - Mechanical Engineering Applicable Safety Assessment Principles

SAP Number	SAP Title	TAG	Assessed Category	WENRA Ref.	IAEA Ref.
EKP - Key Principles					
EKP.1	Inherent safety	T/AST/056	P1-S3	E2.1	2.9 – 2.11 3.2 3.3 3.6 -3.9 4.1 – 4.4
EKP.3	Defence in depth	T/AST/011 T/AST/021 T/AST/056 T/AST/011 T/AST/005	P1-S3	E2.1	2.9 – 2.11 3.2 3.3 3.6 -3.9 4.1 – 4.4
ECS	Safety Classification and standards				
ECS.1	Safety categorisation	T/AST/011 T/AST/056 T/AST/003 T/AST/016 T/AST/057	P1-S3	E3.1	5.1 - 5.3
ECS.2	Safety Classification of structures, systems and components	T/AST/009 T/AST/056 T/AST/016 T/AST/057	P1-S3	G1.1 G2.1	
ECS.3	Standards	T/AST/056 T/AST/003 T/AST/005 T/AST/016 T/AST/057	P1-S3	G2.2 G3.1	3.6
ECS.4	Codes & standards	T/AST/056 T/AST/005 T/AST/016 T/AST/057	P1-S3	C3.1 C3.6	3.6

SAP Number	SAP Title	TAG	Assessed Category	WENRA Ref.	IAEA Ref.
EDR - Design and Reliability					
EDR.1	Failure to safety	T/AST/056 T/AST/016	P1-S3	E9.1	
EDR.2	Redundancy, diversity and segregation	T/AST/036 T/AST/056 T/AST/011 T/AST/003 T/AST/016	P1-S3	E2.1 E9.4 E10.7	2.9 – 2.11
EDR.3	Common cause failure	T/AST/036 T/AST/056 T/AST/016	P1-S3	E10.7	2.9 – 2.11
EDR.4	Single failure criteria	T/AST/011 T/AST/056 T/AST/006	P1-S3	E10.7	2.9 – 2.11
EMT	Maintenance, inspection and testing				
EMT.1	Identification of requirements	T/AST/009 T/AST/056 T/AST/016	P1-S3	K1.1	5.42 6.50 6.81
ELO	Layout				
ELO.1	Access	T/AST/036 T/AST/021 T/AST/009 T/AST/056 T/AST/016	P1-S3		5.43 – 5.44 5.48 5.61 5.65
EPS	Pressure systems				
EPS.1	Removal closures	T/AST/016	P1-S3		
EPS.3	Pressure Relief	T/AST/016	P1-S3		
EPS.4	Overpressure protection	T/AST/016	P1-S3		
EMC - Integrity of metal components and structures					
EMC.5	Defects	T/AST/009 T/AST/016	P1-S3	G3.1	
EMC.7	Loadings	T/AST/016	P1-S3		
EMC.11	Failure Modes	T/AST/056 T/AST/016	P1-S3		
EMC.12	Brittle behaviour	T/AST/016	P1-S3	G3.1	
EMC.22	Material compatibility	T/AST/016	P1-S3	G4.1	
EMC.25	Leakage	T/AST/016	P1-S3		
EMC.26	Forewarning of failure	T/AST/016	P1-S3		

SAP Number	SAP Title	TAG	Assessed Category	WENRA Ref.	IAEA Ref.
EMC.29	Redundancy and diversity	T/AST/009 T/AST/016	P1-S3	E9.4 E10.7	
ECV - Containment and ventilation					6.92 – 6.95
ECV.1	Prevention of leakage	T/AST/021 T/AST/056 T/AST/041	P1-S3	E9.8	
ECV.2	Minimisation of releases	T/AST/056 T/AST/041	P1-S3	E9.8	
ECV.3	Means of confinement	T/AST/021	P1-S3	E9.8 S4.4 S4.5	
ECV.4	Provision of containment barriers	T/AST/021	P1-S3	E9.8 E9.9 E9.10	
ECV.5	Minimisation of personnel access	T/AST/021	P1-S3		
ECV.6	Monitoring devices	T/AST/021	P1-S3		
ECV.7	Leakage monitoring	T/AST/021	P1-S3		
ECV.8	Minimising of provisions	T/AST/021 T/AST/056	P1-S3		
ECV.9	Standards	T/AST/021 T/AST/056	P1-S3		
ECV.10	Safety standards	T/AST/022	P1-S3		

Table A2.2 – Applicable Mechanical Engineering Technical Assessment Guides

No.	Reference No.	Issue	Title
1	T/AST/057	1	Design Safety Assurance
2	T/AST/003	5	Safety Systems
3	T/AST/005	4	Demonstration of ALAEDF AND AREVA
4	T/AST/056	1	Nuclear Lifting Operations
5	T/AST/009	1	Maintenance, Inspection & Testing of Safety Systems, Safety Related Structures and Components
6	T/AST/036	2	Diversity, Redundancy, Segregation and Layout of Mechanical Plant
7	T/AST/022	1	Ventilation
8	T/AST/016	3	Integrity of Metal Components and Structures
9	T/AST/011	1	The Single Failure Criterion
10	T/AST/041	2	Criticality Safety
11	T/AST/021	1	Containment: Chemical Plants