Office for Nuclear Regulation
An agency of HSE

Generic Design Assessment – New Civil Reactor Build

Step 4 Mechanical Engineering Assessment of the EDF and AREVA UK EPR^{TM} Reactor

Assessment Report: ONR-GDA-AR-11-026
Revision 0
11 November 2011

PREFACE

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

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

EXECUTIVE SUMMARY

This report presents the findings of the Mechanical Engineering assessment of the EDF and AREVA UK EPR undertaken as part of Step 4 of the Health and Safety Executive's Generic Design Assessment. The assessment has been carried out on the Pre-construction Safety Report and supporting documentation submitted by EDF and AREVA during Step 4.

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

The scope of the Step 4 assessment was to review the safety aspects of the EDF and AREVA UK EPR reactor in greater detail, by examining the evidence, supporting arguments and claims made in the safety documentation, building on the assessments already carried out for Step 3, and to make a judgement on the adequacy of the Mechanical Engineering information contained within the Pre-construction Safety Report and supporting documentation.

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

This assessment has focused on the safety functions of reactivity control, heat transfer and removal, and containment of radioactive substances, associated with mechanical equipment and systems. In this sense it represents a 'bottom up' assessment, and I have endeavoured to encompass the full range of mechanical items and systems important to safety, albeit subject to the limitations of sampling, to ensure that there are no significant weaknesses in the UK EPR Mechanical Engineering design.

A number of items have been agreed with EDF and AREVA as being outside the scope of the Generic Design Assessment process and hence have not been included in my assessment. These are identified within this report.

For the UK generic design much of the submission has been restricted to the level of high level specifications, for example Stage 1 System Design Manuals, which has limited the extent of my assessment. This restriction specifically applies to information from Factory Acceptance Tests and Site Acceptance Tests, which in general form an important suite of evidential information from a Mechanical Engineering assessment perspective. I recognise that this information is not available within Generic Design Assessment since in many cases suppliers have not yet been selected, and in any case, much of this information is not appropriate for such a generic assessment. However, in order to gain confidence in the Mechanical Engineering design, I have assessed the process described by EDF and AREVA in this respect, discussed examples of such information from other projects, and drawn conclusions accordingly.

I have assessed a broad range of equipment types with important safety functions, including cranes used for nuclear lifting, nuclear ventilation systems, pumps and valves, heat exchangers and associated heat transport systems, Control Rod Drive Mechanisms, and mechanical handling systems. In particular, through undertaking my Step 4 assessment, I have sought to confirm that the equipment described has an adequate nuclear engineering pedigree, is supported by an appropriate degree of Operational Experience Feedback, and has an adequate nuclear safety classification. Where I have identified equipment or processes which I consider to be novel, or which are not aligned to my initial expectations, then I have undertaken a more detailed 'deep slice'

assessment. My assessment of nuclear lifting rigging and load path faults is an example of such a 'deep slice' assessment.

In particular I have identified that for lifts of nuclear safety significance, it is not EDF and AREVA practice to specify the load paths / routes as a design and safety parameter, since they consider the high integrity cranes are not capable of failing. This is not in line with UK regulatory expectations, where cranes are recognised as complex electro-mechanical machines, involving human interaction. A multi-legged safety justification is required covering the mechanical integrity of the crane, the operation of the crane, and taking account of rigging and load path / route faults. EDF and AREVA have now recognised the UK expectations, and have identified preferred load paths (based on As Low As Reasonably Practicable principles) in particular for the Reactor Pressure Vessel head lift, and the lift of the Spent Fuel Pool Stop Gate. I consider these defined load paths represent a significant improvement in safety analyses, and will result in a reduction in the time the lifted load is above the source of recognised nuclear hazards.

I have also taken a particular interest in the safety classification of mechanical equipment. This is part of the graded approach to safety, to ensure that design, procurement, operational, and maintenance attention is focused proportionately on equipment with higher safety importance. In particular, and through Step 4 interactions, EDF and AREVA have now recognised the need to classify duty systems with important safety functions at an appropriate level. These duty systems are the parts of the Nuclear Power Plant which operate under normal conditions, but whose failure is the initiating event for a fault sequence. An example of such a duty system is the main containment polar crane. EDF and AREVA have also now recognised the need to classify mechanical equipment based on the totality of its safety functions, and not simply limited to its pressure boundary containment safety function.

I have also taken a particular assessment interest in the Spent Fuel Cask Transfer Facility, used to transfer fuel out of the spent fuel pool. The UK EPR uses a bottom loading design to avoid the lifting hazard associated with a large cask of spent fuel. This introduces the requirement for a complex interface between the spent fuel cask and the underside of the cask loading pit which is connected to the spent fuel pool. However, this design is being successfully used in the N4 Nuclear Power Plants in France, and is supported by good Operational Experience Feedback. Through my Step 4 assessment, I am now satisfied with the Mechanical Engineering design of this feature, and the associated systems used to isolate the spent fuel pool from the cask loading pit, and thus protect the spent fuel pool from a loss of water, in the unlikely event of a leakage fault.

In some areas where there has been a lack of detailed information Nuclear Directorate will need additional information in Phase 2 (Site Licensing) and these requirements are identified as Assessment Findings to be carried forward as normal regulatory business. These are listed in Annex 1.

An example of an Assessment Finding is that the UK EPR diesel engines and systems do not adequately take into account the required implementation of amendments to regulations in respect of fuels, namely the Motor Fuel (Composition and Content) Regulations 1999. The UK EPR uses diesel engines to provide stand-by power supply capability for the Nuclear Power Plant electrical load requirements. This is a standard feature of Nuclear Power Plants throughout the world. However, the reliability of engine starting and continuity of operation when demanded can be adversely affected by such changes in fuel composition, and this needs to be adequately accounted for in the diesel engine and associated systems design and maintenance arrangements.

In general I have concluded that the UK EPR has evolved from a good nuclear engineering pedigree, and the Mechanical Engineering systems and equipment are well supported by

Operational Experience Feedback. I have not identified any concerns within this report that would require resolution before the Health and Safety Executive would agree to the commencement of nuclear safety related construction of a UK EPR reactor in the UK. Therefore I have not identified any Generic Design Assessment Issues within this report.

Overall, based on the sample undertaken in accordance with Nuclear Directorate procedures, I am satisfied that the claims, arguments and evidence laid down within the Pre-construction Safety Report and supporting documentation submitted as part of the Generic Design Assessment process present an adequate safety case for the generic EDF and AREVA UK EPR reactor design. The UK EPR reactor is therefore suitable for construction in the UK, subject to assessment of additional information that becomes available as the Generic Design Assessment Design Reference is supplemented with additional details on a site-by-site basis.

FOREWORD

Mechanical Engineering

In carrying out this assessment, the term ‘Mechanical Engineering’ encompasses Structures, Systems and Components (SSC) 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 safety function pressure boundaries. Notwithstanding this definition, a number of static components will also be of interest to the Mechanical Engineering discipline, and subject to appropriate assessment.

Examples of SSCs that are considered to be of interest include:

- Control Rod Drive Mechanisms.
- Pumps.
- Valves, (check valves, motor operated valves, safety relief valves, and isolation valves).
- Cranes.
- Mechanical handling systems.
- Nuclear ventilation systems used to augment nuclear containment barriers.
- Heating Ventilation and Air Conditioning (HVAC).
- Diesel generators.

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

- Heat exchangers.
- Gloveboxes, cabinets.
- Stillages.
- Seals.
- Strainers.

Structural Integrity aspects with reference to the containment safety function pressure boundaries and 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

AFCEN	Association Française pour les règles de Conception, de construction et de surveillance en exploitation des matériels des Chaudières Electro Nucléaires
ALARP	As Low As Reasonably Practicable
ASME	American Society of Mechanical Engineers
ASN	Autorité de Sûreté Nucléaire (French nuclear safety authority)
BTS	Book of Technical Specifications
BTR	Book of Technical Requirements
BMS	Business Management System
C&I	Control and Instrumentation
CCWS	Component Cooling Water System
CHRS	Containment Heat Removal System
CMF	Change Management Form
CRDM	Control Rod Drive Mechanisms
CVCS	Chemical and Volume Control System
DNB	Departure from Nucleate Boiling
EBA	Containment Sweep Ventilation System
EBS	Extra Boration System
EFWS	Emergency Feed Water System
EPDM	Ethylene Propylene Diene Monomer
EMIT	Examination, Maintenance, Inspection and Testing
EOT	Electric Overhead Travelling
EQ	Equipment Qualification
ESWS	Essential Service Water System
FA	Fuel Assembly
FAT	Factory Acceptance Tests
FA3	Flamanville 3 Nuclear Power Plant
FLIV	Full Load Isolation Valve
FOAK	First of a Kind
GDA	Generic Design Assessment
HEPA	High Efficiency Particulate Arrestor
HSE	Health and Safety Executive
HVAC	Heating Ventilation and Air Conditioning
IAEA	International Atomic Energy Agency

LIST OF ABBREVIATIONS

IRS	Incident Reporting System
IRWST	In-containment Re-fuelling Water Storage Tank
INPO	Institute of Nuclear Power Operators
I&C	Instrumentation and Control
LC	Licence Condition
LHSI	Low Head Safety Injection
LLW	Low Level Waste
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MCR	Main Control Room
MDEP	Multinational Design Evaluation Programme
MFWS	Main Feed Water System
MHSI	Medium Head Safety Injection
MOV	Motor Operated Valve
MSQA	Management of Safety and Quality Assurance
MSSS	Main Steam Supply System
MSSV	Main Steam Supply Valve
NB	Nominal Bore
ND	Nuclear Directorate (of the HSE)
NEA	Nuclear Energy Agency (of the OECD)
NPP	Nuclear Power Plant
NPSH	Net Positive Suction Head
NVDS	Nuclear Vent and Drain System
OECD	Organisation for Economic Co-operation and Development
OEF	Operational Experience Feedback
OL3	Olkiluoto 3 Nuclear Power Plant
ONR	Office for Nuclear Regulation
PCSR	Pre-construction Safety Report
PLC	Programmable Logic Controller
PMS	Plant Maintenance Schedule
PORV	Power Operated Relief Valve
PSA	Probabilistic Safety Analysis
PSRV	Pressuriser Safety Relief Valve
PWR	Pressurised Water Reactor
RCCA	Rod Cluster Control Assembly

LIST OF ABBREVIATIONS

RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RHRS	Residual Heat Removal System
RO	Regulatory Observation
RPV	Reactor Pressure Vessel
RRC-A	Risk Reduction Category A
RRC-B	Risk Reduction Category B
SAP	Safety Assessment Principles
SAT	Site Acceptance Tests
SDM	System Design Manual
SED	Demineralised Reactor Water System
SFCTF	Spent Fuel Cask Transfer Facility
SFMB	Spent Fuel Mast Bridge
SFP	Spent Fuel Pool
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SIS	Safety Injection System
SQEP	Suitably Qualified and Experienced Person
SSC	Structures, Systems and Components
SSSS	Stand Still Seal System
STUK	The Finish nuclear safety authority
TAG	Technical Assessment Guides
TIG	Tungsten Inert Gas
TQ	Technical Query
UCWS	Ultimate Cooling Water System
US NRC	The United States Nuclear Regulatory Commission
WENRA	Western European Nuclear Regulators' Association
WANO	World Association of Nuclear Operators

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1 INTRODUCTION

1 This report presents the findings of the Step 4 Mechanical Engineering assessment of the UK EPR reactor PCSR (Ref 13) and supporting documentation provided by EDF and AREVA under the Health and Safety Executive's (HSE) Generic Design Assessment (GDA) process. The approach taken was to assess the principal submission, i.e. the Pre-construction Safety Report (PCSR), and then undertake assessment of the relevant supporting documentation on a sampling basis in accordance with the requirements of the Nuclear Directorate (ND) Business Management System (BMS) procedure AST/001 (Ref. 2). The Safety Assessment Principles (SAP) (Ref. 4) have been used as the basis for this assessment. Ultimately, the goal of assessment is to reach an independent and informed judgment on the adequacy of a nuclear safety case.

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

3 A number of items have been agreed with EDF and AREVA as being outside the scope of the GDA process and hence have not been included in this assessment. These are described in Section 2.3.6.

2 NUCLEAR DIRECTORATE'S ASSESSMENT STRATEGY FOR MECHANICAL ENGINEERING

4 The intended assessment strategy for Step 4 for the Mechanical Engineering topic area was set out in an assessment plan (Ref. 1) that identified the intended scope of the assessment and the standards and criteria that would be applied.

2.1 Initial Assessment Plan for Step 4

5 The following table provides a summary of my initial determination of the main elements of the EDF and AREVA safety case in respect of systems containing mechanical equipment.

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

Primary Safety Function	System	Safety Aspect
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 (RCCA), 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 (CRDM) move the RCCAs and enable them to be dropped, to remain as they are, or to be withdrawn.
Reactivity Control	Extra Boration System (EBS)	Emergency addition of Boric acid provides a diverse method of shutting the reactor down.
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.
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.

Primary Safety Function	System	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.
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.
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.
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).
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.
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.

Primary Safety Function	System	Safety Aspect
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.
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.
Containment of radioactive substances	Ventilation Building Containment	The UK 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 inner containment 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 an additional 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 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

Primary Safety Function	System	Safety Aspect
		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 Station 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.
	Ventilation of the Controlled Area of the Operating service centre	The safety function of the ventilation system is to prevent and minimise radioactive releases from the hot laboratories in the Operational 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 Effluent Treatment Building.

Primary Safety Function	System	Safety Aspect
Containment of radioactive substances	Component Cooling Water System	Provide a barrier against leakage of fluid from primary containment and reactor systems.
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 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.

6 Based on this determination of mechanical systems and their high level safety functions, and in conjunction with the work already undertaken during Step 3, I then identified the associated mechanical engineering equipment, and processes for assessment as part of my Step 4 activity.

7 In particular, through undertaking my Step 4 assessment, I have sought to confirm that the equipment described has an adequate nuclear engineering pedigree, is supported by an appropriate degree of Operational Experience Feedback, and has an adequate nuclear safety classification. Where I have identified equipment or processes which I consider to be novel, or which are not aligned to my initial expectations, then I have undertaken a more detailed ‘deep slice’ assessment. My assessment of nuclear lifting rigging and load path faults is an example of such a ‘deep slice’ assessment.

8 My Step 4 plan therefore identified the following initial areas for assessment:

Assessment Area	Description
Design Process: Safety Categorisation and Classification	Verification process for defining a component Safety Classification. Process for identifying safety functional requirements. Ongoing discussions with the RP in respect of the categorisation and classification philosophy.

Assessment Area	Description
Design Process: Transfer of Safety Requirements Through the Project Life Cycle	Assessment of selected System Design Manuals for additional evidence that Responsible Designer role is being retained by the RP. Evidence of the safety functional requirements within the System Design Manuals. Process for transferring the safety functional requirements through the project life cycle. Evidence that the detailed design delivers the required safety functional requirements.
Design Process: Good Engineering Practice	Evidence that the design process encompasses operational experience. Arguments and design criteria for incorporating flexible connections (as a sample).
Design Process: Valve Selection Process	Evidence of an acceptable and auditable process. Process for capturing operational experience. Process for capturing standardisation.
Design Process: Layout / Interfaces	Understanding the role and purpose of the design model throughout the different phases of the project life cycle including the operational phase. The configuration status of the model. The verification process associated with the model. The management and control of systems, discipline and organisational interfaces. The demonstration that the design has sufficient provision for the replacement of mechanical items that are important to safety. Assessment of the replacement sequence of an RCS pump (as a sampled area).
CRDMs	Safety Categorisation and Classification. Evidence from the CRDM trials. Arguments and evidence that support the CRDM equipment classification.
Isolation Valves (containment safety function)	Safety Categorisation and Classification. Identification of safety functional requirements. Arguments and evidence that support the isolation valve equipment classification.
Check Valves	Assessment of the Safety Categorisation and Classification arrangements. Identification of safety functional requirements. Arguments and evidence that support the valve selection and equipment classification process.
Safety Relief Valves	Further evidence in relation to the design and Equipment Qualification issues relating to these Safety Relief Valves, based on their safety classification.
Reactor Coolant System Pump	Safety Categorisation and Classification. Identification of safety functional requirements. Arguments and evidence that supports the RCP equipment classification.

Assessment Area	Description
Cranes	Further evidence in relation to design issues relating to cranes important to safety, based on their safety classification.
Nuclear Ventilation	Further evidence in relation to the design and Equipment Qualification issues relating to nuclear ventilation systems, based on their safety classification. Further understanding of the justification for the design arrangement for iodine filtration in the Main Control Room during the further assessment.
HVAC	Further understanding of the justification of the habitability provision for the Main Control Room under accident conditions.
Gloveboxes / Cabinets	The area will be considered, following a similar approach to the other regulatory areas, although it is anticipated to be of limited interest.
Heat Exchangers	Further assessment activity in this area, with potential attention focused on evidence to support the Equipment Qualification requirements associated with this equipment, based on its safety classification.
Diesel Generator	Safety Categorisation and Classification arrangements. Identification of safety functional requirements. Arguments and evidence that support the diesel selection process and equipment classification.
Spent Fuel Handling	The area will be progressed, following a similar approach to the other regulatory areas of interest, with particular attention focused on the spent fuel route.
Pond Stillages (Fuel Racks)	The area will be considered, following a similar approach to the other regulatory areas of interest specifically: Safety Categorisation and Classification. Identification of safety functional requirements. Incorporation of Operational Experience Feedback.
Radiation Waste Containers	The area will be considered, following a similar approach to the other regulatory areas of interest specifically: Safety Categorisation and Classification. Identification of safety functional requirements. Incorporation of Operational Experience Feedback.
Transportation Flasks	The area will be considered, following a similar approach to the other regulatory areas of interest specifically: Safety Categorisation and Classification. Identification of safety functional requirements. Incorporation of Operational Experience Feedback.
CCWS	Identification of components important to safety that have a reliance on the CCWS. Safety Categorisation and Classification of the CCWS. Identification of safety functional requirements Arguments and evidence that support the CCWS classification.

Assessment Area	Description
Mechanical Filters and Strainers	The area will be considered following a similar approach to the other regulatory areas of interest specifically: Safety Categorisation and Classification. Identification of safety functional requirements. Incorporation of Operational Experience Feedback.

2.2 Standards and Criteria

9 The approach has been to carry out this assessment in accordance with:

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

Those SAPs series considered generally relevant to Mechanical Engineering assessment are listed in Table 1 of this document. Individual SAPs are also detailed within the text of this document against the relevant section.

The Mechanical Engineering assessment has been 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. This approach ensures the assessment provides a targeted, consistent and transparent consideration on the adequacy of the UK EPR design.

10 Generally SAPs capture the requirements of Western European Nuclear Regulators' Association (WENRA) reference levels and the International Atomic Energy Agency (IAEA) Standards Series requirements.

11 It is worth noting, the nature of the Mechanical Engineering discipline generally drives the assessment down to equipment level. Assessment at this equipment level can be extremely wide ranging given the very large number of such items, 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 relating to the selected Mechanical Engineering aspect.

2.3 Assessment Scope

12 The Step 4 assessment scope has been primarily developed from the work undertaken during the Step 3 process, and reviewed and expanded as appropriate through liaison with other assessment disciplines, and as derivative lines of enquiry have emerged through progression of the initially identified assessment scope.

2.3.1 Findings from GDA Step 3

13 At the end of Step 3 of the GDA process good progress had been made in terms of reviewing the EDF and AREVA submission, identifying issues and areas for more

detailed review and discussion, and progressing these to an appropriate level. The Step 3 process and findings are described in detail in the report published in November 2009 (Ref. 6).

14 A degree of confidence was also gained in the design process applied by EDF and AREVA, based on the assessment undertaken in Step 3. However, at that stage the safety function categorisation and equipment classification methodologies did not align with the expectations described in the UK SAPs. Further work was undertaken in this area by EDF and AREVA, which has attracted an appropriate degree of assessment during Step 4, as described later in this report. 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, is considered to be an important area of Mechanical Engineering assessment interest.

15 At the end of the Step 3 assessment process, no Regulatory Observations or Regulatory Issues were identified associated with the EDF and AREVA submission.

2.3.2 Additional Areas for Step 4 Mechanical Engineering Assessment

16 The following additional areas for assessment were identified during the Step 4 process, through liaison with other assessment disciplines, or as derivative areas from previously identified lines of enquiry:

- Containment doors and hatches.
- RPV leak detection system.
- Nuclear drainage systems.
- Medium Head Safety Injection Pump.
- Main Feedwater System.
- Emergency Feedwater System.

2.3.3 Use of Technical Support Contractors

17 No technical support contractors were used to support the Mechanical Engineering assessment of the EDF and AREVA UK EPR reactor design.

2.3.4 Cross-cutting Topics

18 A number of topics are by their nature ‘cross-cutting’ (e.g. Probabilistic Safety Analysis (PSA) and Management of Safety and Quality Assurance (MSQA)), however in addition to these, the project has identified the following ‘cross-cutting’ sub-topics:

- Severe Accidents.
- Categorisation and Classification.
- Examination, Maintenance, Inspection and Testing (EMIT) identification.
- Limits and Conditions.
- Design Change processes.

I have taken a specific technical interest in the subject of safety function categorisation and equipment classification, both to assist in the overall cross-cutting adoption of a philosophy by EDF and AREVA which compares well to that described by the UK SAPs, and also since it interfaces directly with Nuclear Site Licence Condition compliance requirements, as described later in this Step 4 report.

2.3.5 Integration with Other Assessment Topics

19 It is recognised that there are a number of areas where there has been a need to consult with other assessors as part of the assessment process during Step 4. These areas have been overseen by the Project Technical Inspectors in conjunction with Assessment Unit Heads to ensure that potential interactions are captured and that duplicate assessment work is prevented. However, all these dependencies have been 'soft' dependencies such that Mechanical Engineering assessment has progressed and been completed without specific input requirements from these other topics.

20 Coordination with other disciplines has also generally been undertaken as part of the normal assessment process. Given the sampling nature of assessment, this process has proved to be effective and efficient in determining the adequacy of safety cases, and identifying areas of weakness for further resolution.

21 It should be noted that some areas of Mechanical Engineering regulatory interest are electro-mechanical in nature, and specifically the delivery of safety functions may rely on adequate control and instrumentation systems, e.g. nuclear lifting / cranes. Although the general control / protection function of these systems is considered to be a valid area for the Mechanical Engineering assessment discipline initially, where potential regulatory concerns are identified, these are notified to the Control and Instrumentation (C&I) assessment team, who will then take the lead.

2.3.6 Out of Scope Items

22 The following items have been agreed with EDF and AREVA as being outside the scope of GDA, (EDF and AREVA letter dated 30 December 2010, Ref. 36):

- Final nuclear ventilation stack height and associated calculations, (final stack characteristics are site dependent).
- Equipment qualification reports, (documents are strongly linked to the choice of supplier which is outside the GDA scope, although sample information has been provided to illustrate the methodology employed).
- Supplier list of deliverable documents – (supplier lists of documents will vary from one project to the next).
- Heat sink characteristics, (the single sea based heat sink is generic and hence within the GDA scope, however final details of the design and specific sizing of the heat sink are site dependent and hence outside the GDA scope).

23 Furthermore, any Mechanical Engineering features within the UK EPR to support the handling of mixed oxide fuel are outside the scope of GDA. In addition, I have not assessed any features of the design associated with the use and handling of new fuel made from reprocessed uranium.

24 In terms of design documentation, only Stage 1 System Design Manuals are included within the scope of GDA, as described in Section 4.1 of this report covering the Design Process.

3 EDF AND AREVA'S SAFETY CASE

25 A safety case is generally assessed by identifying the claims on structures, systems and components, and people, and then by 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.

26 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 reactivity control, heat transfer and removal, or containment of radioactive substances.

27 The EDF and AREVA PCSR (Ref. 13, 14) used as the basis for this assessment does not collate all information relevant to Mechanical Engineering as a separate topic within the document. I have therefore identified references to mechanical equipment from the appropriate safety case chapters, and pursued my assessment accordingly. The equipment and processes I have selected to assess are reported in the following Section 4 of this report. As a result of there being no large Mechanical Engineering submissions to assess, my assessment has been based on a series of meetings with EDF and AREVA. During these meetings the depth and nature of the Mechanical Engineering design have been tested by questioning, and by the examination of design information, including that from existing EPR projects.

28 This assessment approach has interfaced with the approach adopted by other disciplines, including coordination with the areas of Fault Studies and Probabilistic Safety Assessment, as well as Internal Hazards, to provide a holistic assessment in terms of claims, arguments and evidence covering Mechanical Engineering items important to safety.

4 GDA STEP 4 NUCLEAR DIRECTORATE ASSESSMENT FOR MECHANICAL ENGINEERING

4.1 Design Process

29 I have undertaken a sampled assessment of the EDF and AREVA design process, to ensure they have robust design practices in place that adequately manage interdisciplinary requirements, interfaces, and with the necessary degree of Quality Assurance. I consider this to be an important aspect which underpins the safety justification of the UK EPR design. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment Principle MS.2 (Ref. 4) states 'The organisation should have the capability to secure and maintain the safety of its undertakings.'

30 I also consider from a Mechanical Engineering perspective a significant quantity of evidence is only collated from carrying out the Factory Acceptance Tests (FAT) and Site Acceptance Tests (SAT) (e.g. non-active commissioning tests of equipment, and their integrated system tests). This information is generally not available within GDA, and much detailed design for mechanical equipment is associated with equipment selection and procurement. In recognition of this, my Step 4 assessment has focused on the design specifications, processes, and the transfer of design criteria that are important to safety through to the supply chain, to support the detailed design, procurement and manufacturing phases. As part of my assessment of particular equipment types, which is reported later in this document, I have also assessed examples of equipment types to provide evidence of process and of adequate Operational Experience Feedback. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment Principle EQU.1 (Ref. 4) states '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.'

31 From a GDA perspective I targeted three key areas for evidence that I consider important to confirm EDF and AREVA are an acceptable Responsible Designer, and are able to manage and control a safety important design process; (reference should be made to the technical assessment guide, T/AST/057, (Ref. 7), for explanation of organisational terminology). In particular I consider that EDF and AREVA should have:

- Adequate arrangements in place to transfer the safety functional requirements onto the supplier to allow the detailed design and manufacturing to be carried out.
- Adequate arrangements in place to enable the identification and subsequent transfer of the plant operating limits and conditions to a future licensee, to allow them to undertake their regulatory duties and to generate adequate plant Operating Rules.
- Adequate arrangements for the identification of items important to safety through an appropriate equipment classification process, to allow adequate design and procurement, and generation of a Plant Maintenance Schedule to support the Examination, Maintenance, Inspection and Testing requirements of Nuclear Site Licence Conditions.

The subject of Operating Rules, safety categorisation, classification, and Examination, Maintenance Inspection and Testing (EMIT) is discussed later in Sections 4.2 and 4.3 of this report.

32 In undertaking my assessment, I have used the internal ND technical assessment guide, Design Safety Assurance, T/AST/057 (Ref. 7), to guide my process and conclusions.

4.1.1 Assessment

4.1.1.1 Design Organisation

33 EDF and AREVA described their general design process and their arrangements relating to the development of the Mechanical Engineering design through a typical project life cycle. The information covered EDF and AREVA's organisation framework, the control and management of interfaces, quality assurance and design reviews.

34 The information identified, as an example, a standard organisation framework structure, which is set up as 3 different levels and is the same basis as that for delivering the Flamanville 3 project.

35 The Level 1 organisation is EDF who have prime responsibility for:

- Definition and input data, design, technical and engineering references, acting as Responsible Designer (which includes being the Intelligent Customer), and carrying out the surveillance, verification and acceptance of a supplier detailed design.
- Contractual requirements for Level 2 engineering.
- Surveillance and review of Level 2 engineering activities.
- Overall cross-cutting engineering activities.
- Preparation of documents for plant operations.

36 Level 2 organisations are the delivery design teams that take the defined input data and evolve the design from principles into engineering concepts.

37 Level 2 organisations consist of AREVA, SOFINEL and EDF entities who are responsible for:

- NSSS design.
- Technical specifications of equipment and buildings.
- Safety analyses.
- Technical assessment of bids.
- Preparation of installation documentation.
- Preparation of documentation for the Safety Authority.
- Surveillance, review of engineering performed by Level 3 organisations.
- Reporting of non- conformities to the Level 1 organisation.

38 Level 3 organisations are suppliers who typically deliver the detailed design and manufacture the engineering components ready for installation at site. I noted that AREVA own several organisations that operate at the Level 3 of the framework organisation.

39 I discussed the ability to carry out adequate surveillance and verification on Level 3 organisations. EDF and AREVA stated that an integral part of the supplier's contract is the issue of design documents to them for either information, observation, or for their use

as appropriate. The delivery documents ensure an adequate audit trail and design substantiation exists for equipment to achieve its safety and regulatory requirements and to enable the further transfer of the design intent and safety requirements to site to support installation, commissioning, operations and maintenance phases.

40 Examples of Level 3 suppliers' documents submitted (to Level 2) include:

- Design
  - i) Drawings, calculations, qualification reports, operational and maintenance manuals.
- Manufacturing
  - i) Procedures, specifications, material certification, non-compliance sheets, welding logbooks, control procedures, packing and transportation requirements.
- Factory Acceptance Tests
  - i) Test specifications and reports.
- Site works
  - i) Conformity declaration, installation method statements, test instructions and reports.

41 I consider that the design process and organisation as described provides some adequate evidence of EDF and AREVA satisfactorily demonstrating their Responsible Designer role in terms of generating and controlling technical information of a Mechanical Engineering nature, which has formed the basis of my assessment. This covers aspects of the procurement of equipment from a supplier, who is contracted to develop a concept design through the detailed design phase and onto manufacturing equipment and its release for installation at a site. However, I am aware through liaison with my MSQA assessment colleagues that they have a number of Assessment Findings which relate to the overall design process, organisation, and Quality Assurance arrangements, including control of suppliers and procurement.

42 To date there are a limited number of mechanical items that have been identified to a specific supplier and that are complete in terms of detailed design. As a consequence the ability to seek evidence from detailed design substantiation, and from carrying out the FATs and SATs has been limited. I consider this aspect to be an Assessment Finding (**AF-UKEPR-ME-01**); a future licensee to make available upon request evidence of the detailed design substantiation, FATs information, and SATs information for individual mechanical items and their associated systems, which are important to safety.

4.1.1.2 System Design Manuals

43 System Design Manuals (SDM) have an important role throughout the design life cycle of the Nuclear Power Plant (NPP). They are design control documents, which provide part of the audit trail to how a system has evolved from its initial conception through detailed design, commissioning and the operational stage. In addition they collate engineering evidence that confirms a safety functional requirement is achieved from a Mechanical Engineering perspective. Given the scope of GDA, only Stage 1 SDMs are part of the formal EDF and AREVA submission, although other SDM information has also been provided as examples of design process.

44 Each System Design Manual (SDM) typically consists of 10 parts, which comprise of:

- Part 1; covers the System Design Manual revision history.
- Part 2; covers the system operation, role, design basis, outline system description and system monitoring arrangements.
- Part 3; covers system and component sizing.
- Part 4; covers flow diagrams, functional schematics and detailed design.
- Part 5; covers I&C design, power supplies, functional I&C diagrams, failure modes of system and I&C programming.
- Part 6; covers operator interface design, mimic displays, alarm displays, system operating procedures.
- Part 7; covers component schedule, valves schedule, plant item list, operator command and alarm schedules.
- Part 8; covers wiring and cabling description.
- Part 9; covers references.
- Part 10; covers other documents.

45 The System Design Manuals undergo a three stage process to cover the project life cycle, which comprises of:

- Stage 1; which captures the concept system design.
- Stage 2; which captures the detailed system design.
- Stage 3; which captures the “As Built” system design.

46 My assessment has confirmed that the current status of the SDMs are typically at Stage 1, which I consider is acceptable to support carrying out a meaningful GDA from a Mechanical Engineering perspective.

47 Stage 1 SDMs typically evolve to include the first 5 parts as described above with main inputs being from:

- The system role and design basis.
- Safety classification.
- Functional requirements.

48 Outputs from Stage 1 SDMs are equipment technical specifications, which give consideration to the construction rule requirements. e.g. standards such as RCC-M (Ref. 51), RCC-E (Ref. 52) and those of the American Society of Mechanical Engineers (ASME) etc.

49 Stage 2 SDMs typically evolve to include the first 8 parts as described above with the main inputs being from:

- The developed equipment specifications.
- Feedback from suppliers on proposed equipment to meet the specification.
- Detailed operating studies on equipment integration.

50 Stage 3 SDMs typically evolve to include all 10 parts as described above with the main inputs being from:

- Manufacturing and installation.
- Testing and commissioning.

51 As EDF and AREVA develop their detailed design and their SDMs, further evidence is generated to underpin the equipment safety claims and arguments. The Part 4 System Design Manual collates and presents the system flow diagrams for the relevant design stage. As the detailed design evolves, all the mechanical drawings are collated within Part 4 of the System Design Manual.

52 I have undertaken a sample assessment of System Design Manual; Safety Injection System (SIS) and Residual Heat Removal (RHR) Part 2 - System Operation (Ref. 15). I consider this provides good evidence of the safety requirements, considerations, assumptions and dependencies for the system and the system role to support the following safety functions during normal operations, design basis accidents and in a severe accident scenario:

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

53 I have undertaken a sample assessment of System Design Manual; Safety Injection System (SIS) and Residual Heat Removal (RHR) - Part 3 - System Sizing (Ref. 16). I consider this provides good evidence of the design safety requirements, operating parameters, assumptions and system characteristics which are specified to allow the detailed design of the system and its components to be developed.

54 I have undertaken a sample assessment of System Design Manual; Extra Boration System (EBS) – Part 2; System Operation (Ref. 17). I consider this provides good evidence of the safety requirements, considerations, assumptions and dependencies for the system and the system role to support the following safety functions during normal operations, design basis accidents and in a severe accident scenario:

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

55 I have undertaken a sample assessment of System Design Manual; Extra Boration System (EBS) - Part 3; System and Component Sizing (Ref. 18). I consider this provides good evidence of the design safety requirements, operating parameters, assumptions and system characteristics being specified to allow the detailed design of the system and its components to be developed.

56 I have also undertaken a sample assessment of Equipment Specification for the Control Rod Drive Mechanisms (Ref. 19). I consider this provides good evidence that EDF and AREVA act as a Responsible Designer throughout the procurement of an item important to safety. For example there is good evidence that EDF and AREVA define the appropriate safety functional requirements and there is a requirement for EDF and AREVA to endorse particular aspects of the detailed design, design substantiation, manufacturing, assembly and the acceptance tests.

57 During the assessment, the safety function categorisation and equipment classification methodology has developed to align more closely to UK expectations (Section 4.2), and the SDMs I have sampled do not reflect this. I therefore consider it to be an Assessment Finding (**AF-UKEPR-ME-02**) that all SDMs are required to be reviewed and revised appropriately to align with the revised UK EPR safety categorisation and classification methodology, which is an accepted outcome of the work to resolve cross-cutting Regulatory Observation RO-UKEPR-043, (Ref. 11). The GDA project has also raised cross-cutting GDA issues **GI-UKEPR-CC-01** and **GI-UKEPR-CC-02** which relate to this area. The complete GDA Issues and associated actions are formally defined in Annex 2 of the UK EPR Cross-cutting Assessment Report ONR-GDA-AR-11-032 (Ref. 47).

58 Notwithstanding this Assessment Finding, I am satisfied with the descriptions of the SDM documentation and associated process from a Mechanical Engineering GDA perspective.

4.1.1.3 3D Model

59 EDF and AREVA described their 3D design model, which covered the model's:

- Role and purpose throughout the various stages of the project life cycle.
- The output deliverables.
- Design status and configuration.

60 The model is utilised to manage the various discipline interfaces, to achieve adequate design management control and an appropriate audit trail. The model architecture is split into 4 specific areas:

- Discrete individual study areas.
- Common area (no validation).
- Validation area.
- Validated area.

61 Design Engineers within the discrete individual areas carry out studies on particular aspects and within their discipline topics. Studies are completed in these areas and then the information is transferred into the common area. Examples of these activities carried out in the discrete areas include: pipework routing, electrical cable tray routing and supports, HVAC and nuclear ventilation equipment and ducting routing, civil works, I&C cabling and cabinet studies. On carrying out an individual study it is possible to interface with other design packages, for example, carrying out a stress analysis on a length of pipe.

62 Within the study areas the design evolves, taking into account interfaces, other disciplines and system requirements etc. Once a design study is sufficiently developed it is transferred into the common area of the model (at this stage the model is not validated). The model is split into many different areas and these areas are individually owned and the ability to make changes is also controlled within the disciplines. When an area is ready to be validated it is then the subject of a multidisciplinary review that considers all the necessary aspects such as interfaces with adjacent areas, system requirements etc.

63 In response to questions through discussion, EDF and AREVA stated that the management of interfaces is the responsibility of the project department.

64 During discussion at the October 2009 Technical Meeting a number of sample documents were presented from the Flamanville 3 (FA3) Nuclear Power Plant (as examples of process) to clarify the validation review process and the auditable trail.

65 Once an area has been validated the individual area in question is transferred into the validated area of the model. From this area of the model it is possible to generate a procurement and installation package. The model is able to automatically produce validated isometric drawings, material take offs, ventilation drawings, support drawings, and layout and general arrangement drawings for the Mechanical Engineering aspects.

66 In response to questions, EDF and AREVA explained the role of the ‘Validator’, who signs off the process within a system, and is a Suitably Qualified and Experienced Person (SQEP) nuclear engineer.

67 EDF and AREVA also stated that for some disciplines, specifically pipework, the process operates on a closed loop principle whereby information from the suppliers’ 2D drawings is fed back into the 3D design model. For other disciplines, e.g. ventilation, the process is one way only, with information being taken from the 3D design model and issued to the suppliers.

68 In response to my questions, EDF and AREVA stated that part of the piping design process involves design teams checking the adequacy of piping support brackets, based on the loads provided by the separate piping stress design team. The design philosophy is to use a catalogue of standard piping support arrangements.

69 The overall process uses design freeze arrangements, which are built into the design flow and configuration process.

70 The 3D design model has the ability to capture the project ’as built’ status and to take into account design changes. It is therefore potentially not only a useful tool for supporting operations but a useful aid during the final decommissioning phase. In response to questions, EDF and AREVA made the general statement that if the equipment is installed to within the construction tolerance, then no changes are made to the model to reflect the ‘as-built’ status. If installation is made outside of tolerance, then this would invoke the design change process. I consider this to be a rational approach, and in line with my expectations.

71 The design arrangements also include a Design Review process, which is separate, and in addition to the 3D design model validation process, and is effected via a meeting, including suitable attendance from operators. EDF and AREVA tabled the output notes from such a meeting during discussions, which I briefly reviewed and considered to be acceptable.

72 In response to additional questions, EDF and AREVA stated that they used the 3D design model for maintainability studies. They also explained that the model links to erection / construction sequencing requirements, and that the design configuration is standard for French Nuclear Power Plants.

73 The information provided explanation and evidence that a satisfactory design process is in place, is being followed for a typical project, and managed with reference to building and equipment layouts. I consider that EDF and AREVA were able to supply good answers to my questions, and tabled sample evidence which was in line with my expectations. From a GDA perspective, I consider that appropriate processes are in place for the management of multi-disciplinary interfaces for the various systems and organisations within the overall design team, through use of the 3D model.

74 In summary, I consider that EDF and AREVA have described an adequate design process regarding use of their 3D model, and I am satisfied from a GDA perspective.

4.1.1.4 Design Change Process

75 EDF and AREVA described their Design Change process. Through discussion, they confirmed that the design submitted for GDA only goes down to Stage 1 level of SDMs, and that lower levels of documentation will be dependent on site specific procurement arrangements.

76 EDF and AREVA stated the design change process for an actual UK build would be developed along similar lines as that used for the GDA process itself.

77 In response to questions they stated that the sentencing of design changes is undertaken on a committee basis.

78 EDF and AREVA then described the basis of the GDA design as being the SDMs for FA3, plus the Change Management Forms (CMF) raised and applied to this project; plus the CMFs which will be UK specific.

79 I have not pursued this aspect of the design process to any significant depth, due to the sampling nature of my assessment, and I am also aware that my colleagues in the discipline of MSQA have undertaken further work in this area. However, from my limited work in this area, I did not identify any significant concerns and so I consider the adopted design change process is generally in line with my expectations. The GDA project has also raised Cross-cutting GDA issue **GI-UKEPR-CC-02** (Ref. 47) which relates to this area.

4.1.1.5 Equipment Qualification

80 EDF and AREVA described their processes in relation to Equipment Qualification (EQ). I consider the information covered a narrow definition of EQ in the sense of gaining assurance that equipment (including both mechanical and electrical equipment) is able to deliver its safety function through degraded service conditions through its anticipated lifetime, as opposed to looking at the overall Mechanical Engineering safety functionality of the equipment. The specific phenomena associated with the EQ processes described are:

- Radiation exposure.
- Seismic.
- Ageing.
- Temperature, humidity and pressure.

81 On this basis, I consider that EQ represents a necessary but not sufficient process for mechanical equipment in terms of providing assurance in respect of safety function. I consider assurance relating to the other aspects is derived from the safety case justification, design process integrity, FATs and SATs, Operating Limits and Conditions, and future plant Examination, Maintenance, Inspection and Testing.

82 Methodologies are employed as part of the process, covering laboratory analysis, ‘by analogy’ assessment, and by Operational Experience Feedback (OEF) (albeit not used alone). EDF and AREVA described the different qualification practices embodied in the

German, French and US rules / standards, and stated that a comparative study carried out in the 1990s had determined that the methodologies are effectively equivalent in terms of objective, and practice from a macroscopic testing perspective.

83 Measures are undertaken to preserve EQ during manufacture, erection on site, and during operation. This includes selection and surveillance of suppliers against products, Qualification Preservation Sheets to preserve EQ during site erection (with identification of various witness points), and analysis of OEF during operation.

84 Documentation and process associated with EQ for the UK EPR comprise:

- A Qualification Strategy developed by the supplier, including which codes will be used, (noting that for coherency codes cannot be combined, i.e. by using parts from different codes).
- Qualification Specifications for any identified testing, describing the tests, measured parameters, and stating acceptance criteria.
- Qualification Tests / Analyses Reports, describing the results of the EQ undertaken.
- A Qualification Summary Report.
- Qualification Preservation Sheets, highlighting key points for qualification preservation throughout equipment lifetime.
- A Qualification File, to gather together all the appropriate references associated with EQ.

85 Although EDF and AREVA do not specify the qualification methods (where there are options within the controlling documents), they retain the right to reject the supplier's proposed methodology.

86 Part of their EQ process is to assess whether manufacturers would be able to produce series equipment (i.e. effectively equivalent) for the 60 year plant design life, in terms of identifying the possibility of obsolete technology etc; noting that this assessment does not cover commercial / business aspects. EDF and AREVA explained that they do have separate supply chain processes to cover for business aspects.

87 Circa 90% of qualification of electrical equipment is done by testing; (electrical equipment is not generally qualified by analogy).

88 The supplier undertakes bounding / grouping of equipment into standard conditions, including seismic demand. However, for reasons of interchangeability, EDF and AREVA may also have undertaken some bounding of seismic demand, prior to the specification having been given to the supplier.

89 The EQ process does include for EMIT related ageing surveillance. The EQ process also specifically covers the whole equipment system / chain associated with delivery of a safety function, e.g. covering pumps, actuators, electrical cables etc, to ensure that there is no weak link in the system.

90 In response to questioning as to whether spray effects from plant failures have been considered as a qualification requirement, EDF and AREVA stated that this is addressed through diversity and segregation of equipment, such that consequential failure is limited to one division; where trains are not physically segregated, then barriers are introduced. I consider this response to be reasonable from a Mechanical Engineering perspective, although this subject falls generally into the subject area of Internal Hazards assessment.

91 They also confirmed that there is a process for assessing and sentencing derogations requested as part of the EQ testing process. EDF and AREVA stated that generally only unused items identical to those tested are accepted for subsequent plant use, although they do have a system for allowing specific large items (e.g. pumps, valves) to be used on plant following seismic testing for example. These items are designed to meet the seismic demand, and generally do not exhibit damage during testing; (appropriate verification is performed before use of such equipment to ensure that no damage has occurred during testing).

92 For the UK EPR, EDF and AREVA stated that the EQ process will need to accommodate new equipment and suppliers, the licensee organisation and the overall UK context.

93 In response to a technical meeting action EDF and AREVA issued a number of design reports for my sample review.

- One report presents the general test specification defining the procedures and terms that are to be applied for test based qualification of valves, for accident operating conditions (Ref. 37). This document is generally aligned with my expectations.
- A further report presents the general test conditions to be applied in the qualification of a pump assembly mechanical seals for accident conditions (Ref. 38). This design report presents various qualification methods applied to mechanical seals, i.e. test based, correlation based, and feedback based qualification, and is generally aligned with my expectations.
- A further report describes the process and evidence required to qualify a pump assembly for accident conditions (Ref. 39) and is generally aligned with my expectations.

94 I noted from my assessment that not all documentation has been produced to date to define all necessary Equipment Qualification requirements. I consider it to be an Assessment Finding (**AF-UKEPR-ME-03**) that a future licensee is required to generate appropriate evidence that Equipment Qualification is adequately specified for all mechanical items important to safety, accounting for new suppliers and the overall UK context.

95 I consider that this area will be of continuing regulatory interest as normal business for Phase 2 of the new nuclear build programme in the UK. However I consider that EDF and AREVA have described a reasonable and rational process, which aligns with SAP EQU.1, and I am satisfied from a GDA perspective.

4.1.2 Findings

AF-UKEPR-ME-01: *The licensee shall make available evidence of the detailed design substantiation, Factory Acceptance Test (FAT) information, and Site Acceptance Test (SAT) information for individual mechanical items and their associated systems, which are important to safety. Target Milestone – fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-02: *The licensee shall ensure that all System Design Manuals (SDM) are reviewed and revised appropriately to align with the UK EPR safety categorisation and classification methodology, which is an accepted outcome of the work to resolve cross-cutting Regulatory Observation RO-UKEPR-043. Target*

Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.

AF-UKEPR-ME-03: *The licensee shall generate appropriate evidence that Equipment Qualification is adequately specified for all mechanical items important to safety, accounting for new suppliers and the overall UK context. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

4.2 Safety Function Categorisation and Equipment Classification

96 Safety function categorisation and associated equipment classification are important considerations from a Mechanical Engineering perspective, although this topic area is cross-cutting in that it affects the range of assessment disciplines to a greater or lesser degree. I consider the following Safety Assessment Principles to be relevant to this aspect:

- Safety Assessment Principle ECS.1 states '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.'
- Safety Assessment Principle ECS.2 states '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 regards to safety.'

97 It is for EDF and AREVA to generate their own structure to reflect the principles described above, based on considerations of hazard and risk, but this subject is important for mechanical equipment since it is an input to the definition of design requirements, procurement processes (specifically assurance activities), installation and commissioning activities, and of particular importance the Examination, Maintenance, Inspection and Testing (EMIT) requirements which are regulated during plant operation under Nuclear Site Licence Condition 28.

4.2.1 Assessment

98 The subject of safety function categorisation and equipment classification has been raised as a cross-cutting Regulatory Observation onto EDF and AREVA, RO-UKEPR-043 (Ref. 11). I have supported the generation of the observation and actions within this regulatory observation, and have participated in subsequent meetings to progress the resolution of the subject. Additionally the subject has been discussed as a regular agenda item at the Mechanical Engineering technical meetings which have been held as an important part of my assessment process.

99 EDF and AREVA had initially described a generally rational and systematic process to address this area within the design and safety justification process; but it did not align well with the expectations described by the UK Safety Assessment Principles (SAP) in that it did not clearly differentiate between safety function categorisation and equipment classification. Furthermore, this initial system did not explicitly assign equipment classification to many major items of mechanical equipment (in terms of their Mechanical

Engineering safety functionality), which I considered to be a significant shortfall since it would not provide the key safety case link to subsequent EMIT requirements (to be regulated under Nuclear Site Licence Condition 28). In terms of classifying these major items of mechanical equipment, I also considered it appropriate that 'duty' system equipment, (referred to as Safety Related Systems within the ND Technical Assessment Guides, (Ref. 7)), are classified at an appropriate level. These 'duty' systems represent the normal operational equipment used within a Nuclear Power Plant, but which have important safety functions (i.e. reactivity control, heat transfer and removal, and containment of radioactive substances), and whose failure is typically the initiating event within a fault sequence. An example of such a 'duty' system is the main containment Polar Crane.

100 EDF and AREVA have progressed this subject, and responded positively to the guidance which has been provided, and have generated a report to reflect the application of their new methodology, Classification of Structures Systems and Components, (Ref. 32). This document illustrates the fact that although EDF and AREVA have retained the architecture of their pre-existing structure to define the system categorisation requirements, they have now recognised the structure described in the UK SAPs, and have generated an additional layer of information to define safety function categorisation and equipment classification, which in my judgement compares well to UK SAPs (ECS.1 and ECS.2).

101 I also note that EDF and AREVA have recognised the expectation to classify 'duty' system equipment as appropriate, and some work in this respect is clearly shown in this latest document, (Ref. 32). For example the nuclear lifting equipment has now attracted an appropriate level of nuclear safety classification. Furthermore, EDF and AREVA have also recognised the need to classify mechanical equipment based on the totality of its engineering functionality, and not simply limited to its pressure boundary containment safety function. However, at the time of writing this Step 4 report the exercise is not yet complete, and I consider it to be an Assessment Finding (**AF-UKEPR-ME-04**) that this work should be undertaken and finalised. The GDA project has also raised cross-cutting GDA issue **GI-UKEPR-CC-01** (Ref. 47) which relates to this area.

4.2.2 Findings

> **AF-UKEPR-ME-04:** *The licensee shall ensure that safety function categorisation and equipment classification is specified for all mechanical items important to safety, specifically including equipment which is the source of postulating initiating events (i.e. safety related systems, also termed duty systems). I consider that initially this exercise should focus on the major items of mechanical equipment, at an appropriate level to reflect the GDA workscope. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

4.3 Limits and Conditions and EMIT Identification

102 A key feature of a safety case is the identification of the limits and conditions which define the safe operating envelope for plant operation. In the UK these limits and conditions are termed Operating Rules and are regulated through Nuclear Site Licence Condition 23. Although making and implementing the arrangements associated with Licence Condition

23 (LC 23) is the responsibility of a future licensee, it is important that sufficient information has been generated through the safety case documentation suite to facilitate this, and it is also important that the Responsible Designer recognises this requirement, and will in future support a licensee by providing appropriate technical information and support.

103 In a similar fashion, a key feature of a safety case is also the identification of Examination, Maintenance, Inspection and Testing (EMIT) requirements for the Structures, Systems and Components (SSC) within the Nuclear Power Plant. In the UK these are regulated through Nuclear Site Licence Condition 28 (LC 28), which includes the requirement to generate a Plant Maintenance Schedule to define the safety important maintenance activities, with appropriate periodicities. I consider the following Safety Assessment Principles to be relevant to this aspect:

- Safety Assessment Principle SC.6 (Ref. 4) states 'The safety case for a facility should identify the important aspects of operation and management required for maintaining safety'.
- Safety Assessment Principle EMT.1 (Ref. 4) states 'Safety requirements for in-service testing, inspection and other maintenance procedures and frequencies should be identified in the safety case.'

4.3.1 Assessment

104 As part of my Mechanical Engineering assessment, I have reviewed the accessibility and practicability for maintenance of mechanical equipment on a sampled basis as I have discussed various equipment types, and also discussed the typical maintenance which is undertaken during the lifetime of the NPP. This is described as appropriate in the sections of this report covering particular items of equipment.

105 On discussing the topic of the plant operating limits and conditions at a technical meeting EDF and AREVA stated that Chapter 18 of the PCSR covers the topic in a general sense. However, initial assessment and further discussions on the subject of operating technical specifications failed to achieve an adequate understanding of EDF and AREVA's arrangements.

106 Through subsequent discussions with EDF and AREVA, and liaison with my assessment colleagues, a cross-cutting Regulatory Observation was raised to cover this subject area, RO-UKEPR-055 (Ref. 11), to require EDF and AREVA to develop a coherent and consistent philosophy across all assessment disciplines. I have also attended technical meetings to further discuss the detailed expectations within this Regulatory Observation.

107 EDF and AREVA have now produced an initial response to this RO, EPR Design Basis Limits and Development of Plant Operating Limits and Maintenance Schedules, dated 17th December 2010, (Ref. 20). I have reviewed this document from a Mechanical Engineering perspective, and I am satisfied that EDF and AREVA have recognised the importance of identifying this information for LC 23 and LC 28 compliance as a necessary part of the safety case production process, and they have identified sufficient examples of information and description of process to provide confidence from a GDA perspective, and comparison to SAPs SC.6 and EMT.1. I consider it to be an Assessment Finding (**AF-UKEPR-ME-05**) that this process should be continued to cover all mechanical items important to safety.

108 I am also satisfied that EDF and AREVA have recognised that there is a close link between this subject, and the ongoing work in respect of safety function categorisation and equipment classification.

4.3.2 Findings

> **AF-UKEPR-ME-05:** *The licensee shall ensure that the identification of plant limits and conditions, and EMIT requirements, from the safety case is completed to cover all Mechanical Engineering equipment important to safety. The licensee shall generate sufficient safety case information to satisfy the requirements of LC 23 and LC 28, and specifically a suitable interface shall be established to facilitate transfer of this information from the Responsible Designer, in due course. Target Milestone – fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

4.4 Good Engineering Practice

109 I have decided to assess a number of aspects of the UK EPR against my expectations in relation to Good Engineering Practice. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment Principle EKP.1 (Ref. 4) states 'The underpinning safety aim for any nuclear facility should be an inherently safe design, consistent with the operational purposes of the facility.'

110 I have specifically selected to sample the following aspects against my expectations of Good Engineering Practice in the nuclear engineering context, during my Step 4 assessment:

- Use of Stellite™ within Mechanical Equipment.
- Flexible Connections / Hoses.
- Mechanical Locks / Interlocks.
- Pipework Dead Leg Phenomena.

4.4.1 Assessment

4.4.1.1 Use of Stellite™ within Mechanical Equipment

111 The transport of cobalt atoms into fluid systems through either wear, maintenance dressing of sealing surfaces, or corrosion, is a known problem in Nuclear Power Plants, which can lead to high worker dose rates, through the activation of cobalt due to neutron flux within the primary circuit. However, Stellite™, a cobalt chromium alloy, has very favourable mechanical characteristics leading to its use in valve seats, where there is an onerous mechanical duty. I decided to assess and gain an understanding of the EDF and AREVA strategy to manage and limit the use of Stellite™ within the NPP design.

112 EDF and AREVA provided information covering material selection, with a specific focus on the development activity associated with replacement of Stellite™ as a hard facing material for valve seats, with alternatives which do not contain cobalt, TQ-EPR-1444 (Ref. 10). EDF and AREVA stated that they have a design principle of not using Stellite™ on valves that are positioned within process lines that are in contact with

primary coolant as historically 5% of contamination was due to cobalt 60 (activated from cobalt 59), coming from valves. This is due to the cobalt tending to dissolve into solution, which then becomes activated. This is particularly an issue on valves that are opened routinely, which therefore see mechanical wear. Stellite™ has traditionally been used as the seating material due to its hardness and favourable wear characteristics.

113 Valve hard facings are subject to the following phenomena which can lead to the transport of cobalt atoms into solution:

- Corrosion.
- Thermal shocks.
- Friction and wear.
- Contact pressure.
- Mechanical shocks.
- Fluid cavitation erosion.

114 In order to allow for these effects, the following parameters are important when assessing alternative materials:

- Chemical composition.
- Crystallography.
- Metallurgy.
- Hardness.
- Tensile strength.
- Toughness.
- Wear characteristics.
- Corrosion and pitting corrosion.
- Stress corrosion.

115 EDF and AREVA described various alterative materials to Stellite™ (for example Antinit Dur 300 and NOREM™), and explained that it was for the valve supplier to propose the final selected material, which would be subject to necessary project approval. EDF and AREVA described the impact of the desire to eliminate Stellite™ on equipment qualification, noting that (with few exceptions) Stellite™ is forbidden for valves in contact with the primary circuit fluid, and that suppliers are free to choose the alternative cobalt free hardfacing and associated qualification.

116 In particular EDF and AREVA stated that although NOREM™ is not as good a material from a mechanical perspective as Stellite™, it is one of the best alternatives, and can be used in certain specific locations.

117 They also stated that:

- Some valve selection changes have been implemented to make the associated hardfacings more amenable to the use of NOREM™ (for example), specifically increasing the use of globe valves, as opposed to gate valves.

- Only three valves in contact with the primary circuit utilise Stellite™ hardfacing seats, and these are associated with fault conditions, so contact fluid is not routinely transported around the primary circuit.
- They have considered knowledge, experience and development work undertaken from across the world re Stellite™ elimination / reduction, and not simply limited to their own development programmes.

118 I consider EDF and AREVA provided an acceptable level of information in this area, providing a clear description of optioneering and design improvements to minimise generation of radioactivity within the primary circuit, in comparison to SAP EKP.1. In summary, I am satisfied with EDF and AREVA's approach to minimisation of cobalt in valves, but consider that an Assessment Finding (**AF-UKEPR-ME-06**) is appropriate to ensure that attention remains focused on Stellite™ reduction, including development of new alternative materials, as the overall NPP project progresses.

4.4.1.2 Flexible Connections / Hoses

119 When comparing a section of pipework to a flexible hose, a flexible hose is of a weaker design principle with lower integrity, with containment properties being of a lower reliability; and their duty typically necessitates an increase in human interactions. I therefore decided to assess the EDF and AREVA design requirements and criteria for the use of flexible hoses within the UK EPR NPP design.

120 In response to my questions, EDF and AREVA described their considerations for incorporating flexible hoses within the UK EPR design, TQ-EPR-094 (Ref. 10). There are a number of applications where flexible hoses are utilised, which include:

- Temporary hoses to support air operated maintenance tools.
- Temporary hoses to aid filling and draining of mobile devices.
- Temporary hoses to aid flushing of radioactive contamination process lines.
- Temporary hoses to drain lines to sumps.
- Permanent hoses for de-coupling loads between components and associated process pipework.

121 The Olkiluoto 3 (OL3) NPP design has been reviewed and many flexible hoses have been eliminated by design within the FA3 NPP design, (and hence the UK EPR design). However, EDF and AREVA still anticipate the use of a number of flexible hose connections for the UK EPR NPP design.

122 It is evident from progressing my assessment that there are occasions when flexible hoses are the preferred design choice. An example of this is the use of a flexible hose to uncouple the load on a valve (PTR7135VB) leak detection system. I consider this type of application is an acceptable use of this design choice.

123 EDF and AREVA explained that for the UK EPR flexible hoses are used for 56 applications within the Reactor Building, 31 on radioactive systems and 25 on non radioactive systems. There are 43 applications outside of the Reactor Building, 25 being on radioactive systems and 18 on non radioactive systems.

124 A second specific example where a permanent flexible hose is utilised is on the sampling line of a system drain and vent line. EDF and AREVA advised that the UK EPR design incorporates this arrangement in 3 locations within the Fuel Building.

125 EDF and AREVA described a further case for the temporary use of flexible hoses to enable connections to specific tools and services to allow periodic testing of the Spent Fuel Cask Transportation unit.

126 Discussions identified that the length of flexible hoses is also constrained by an appropriate design rule.

127 On balance I judge all the above examples are acceptable uses of flexible hoses. I am satisfied with the principle and application of this equipment from a GDA perspective in comparison to SAP EKP.1.

4.4.1.3 Mechanical Locks / Interlocks

128 My assessment also targeted the requirement for an NPP design to include adequate provision for the isolation of process lines to enable Examination, Maintenance Inspection and Testing of plant and equipment to be carried out in a safe manner. I therefore assessed EDF and AREVA's design principles and rules for mechanical isolations, locking devices and interlocks, as an important aspect of Mechanical Engineering to support this requirement. Such features are also important to ensure that correct plant line-up is maintained for normal operation.

129 EDF and AREVA described their principles and rules utilised to define adequate isolation requirements on an individual system process line, in response to my line of enquiry regarding mechanical locking devices.

130 Discussions indicated that the utilisation of locks / interlocks follows a number of design principles and rules. Examples of the principles include:

- A manual valve that is required to be maintained in a particular safety position has an integrated mechanical locking device, which may interlock with other valves.
- Valves that are utilised for periodic testing, or infrequent operations (e.g. interconnection of trains) are equipped with a position indication limit switch and if necessary a mechanical lock / interlock.
- Valves that are only used for maintenance incorporate a mechanical locking device and where necessary an interlock.
- The keys supporting the mechanical interlocks are typically under the control of the shift leader within the main control room.

131 In response to questions EDF and AREVA stated that the UK EPR interlock devices are of an established pedigree design principle, and are of a similar design principle to those utilised within the KONVOI (German) NPPs.

132 I am satisfied that EDF and AREVA have demonstrated their design process gives adequate consideration to the principles of mechanical locking devices and interlocks within the UK EPR design. I consider that they are also aligned to the principles expressed in HSG253, the safe isolation of plant and equipment (Ref. 45). I am therefore satisfied with this line of enquiry from a GDA perspective, in comparison to SAP EKP.1.

4.4.1.4 Pipework Dead Leg Phenomenon

133 In recognition that the ‘dead leg’ phenomenon can have detrimental effects on SSCs important to safety, I targeted my assessment in this area to understand how EDF and AREVA’s design process takes this into account; (a ‘dead leg’ is a section of pipework containing fluid which is not subject to normal process flow).

134 EDF and AREVA described the ‘dead leg’ phenomenon in the UK EPR design, and how Operational Experience Feedback has led to a greater understanding of the phenomenon, and hence design changes to avoid and reduce the associated detrimental effects, TQ-EPR-1450 (Ref. 10).

135 A ‘dead leg’ is defined as being an auxiliary system of pipework connected to the Main Reactor Coolant pipework, in which there is no movement of fluid due to isolation by valves. EDF and AREVA stated that historically the phenomenon had not been sufficiently well understood, and the penetration of the turbulent hot fluid into the ‘dead leg’ had been underestimated. Previously the design had assumed that the hot fluid only penetrated by 3 times the pipe diameter into the auxiliary pipe, with the rest of the pipe assumed to be uniformly cold.

136 Operational Experience Feedback from the French fleet of NPPs has indicated a hot fluid temperature far beyond the 3 times the pipe diameter assumed by the design process, and subsequently instrumentation campaigns were set up to better define the thermal hydraulic behaviours in the ‘dead legs’.

137 Based on this improved understanding, EDF and AREVA explained that a ‘dead leg’ can be considered as comprising two distinct parts:

- A part directly connected to the primary loop which is not capable of being isolated, before the first isolation valve.
- A part connected to the auxiliary system, which is capable of being isolated by the use of existing valves.

138 The flow in the primary circuit is now understood to induce turbulent flow in the non-isolable section of pipework to a depth of circa 18 times the inside diameter of the pipe, with the temperature of the turbulent penetration being close to that of the primary pipework. In response to questions, EDF and AREVA explained that the main detrimental phenomenon associated with this turbulent flow penetration is thermally induced fatigue for the first isolation valve, if it is situated within this turbulent penetration zone. Another detrimental phenomenon is accelerated corrosion where a two phase regime is induced (in the adjacent part that is capable of being isolated). Other detrimental material effects such as accelerated metallurgical ageing are also possible due to higher than predicated temperatures.

139 The UK EPR takes into account this improved understanding of the ‘dead leg’ phenomenon by the following design approaches:

- Locating the first isolation valves far enough from the primary pipe location in order to limit temperature variations in the isolatable section of pipework; which avoids the risk of two phase media generation in this pipework section and hence reduces the risk of corrosion associated with this effect. This also avoids thermal fatigue cycling at the first isolation valve location.

- Avoiding locating the end of the turbulent penetration at a pipework transition, e.g. at an elbow, in order to control and define the length of turbulent penetration, and therefore limit higher temperature detrimental effects.

140 I consider that EDF and AREVA described a rational design improvement process based on Operational Experience Feedback, and I am satisfied with the improvements described from a Mechanical Engineering and a GDA perspective in comparison to SAP EKP.1.

4.4.2 Findings

AF-UKEPR-ME-06: *The licensee shall review and consider alternative materials to Stellite™ for applications within the NPP domain, and generate evidence to ensure that material selection is ALARP for the UK EPR in respect of the use of Stellite™. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components – delivery to site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

4.5 Control Rod Drive Mechanisms

141 Control Rod Drive Mechanisms (CRDM) have an important safety function of controlling the nuclear core reactivity within an NPP. Against the background that CRDMs are of an established principle of design with significant operational experience within NPPs around the world, my assessment philosophy during Step 4 has focused on design improvements, associated claims, arguments, and evidence, plus the overall CRDM safety categorisation and classification. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment Principle ECS.3 (Ref. 4) states '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.'

142 During Step 4 I have also assessed the arguments and justification supporting the implementation of design change CMF-013 (Ref. 48), and associated evidence that the safety functional requirements have been substantiated from undertaking the endurance testing. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment Principle EQU.1 (Ref. 4) states '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.'

143 My previous Step 3 assessment considered the CRDM latch assembly as being a particular item important to safety and it should therefore be classified accordingly. Initial assessment of the safety documentation had not addressed this aspect to my satisfaction. To progress my Step 4 assessment I have continued to target the evidence to support adequate classification of the CRDMs. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment Principle ECS.2 states '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 regards to safety.'

4.5.1 Assessment

4.5.1.1 Design and Safety Functions

144 The CRDM's operational functions support:

- Insertion and withdrawal of the 89 Rod Cluster Control Assemblies (RCCA).
- Holding the RCCA at a selected position within the core.
- Indication of the RCCA position within the core.

145 The CRDM's safety functions are to:

- Manage the core reactivity, during reactor trip.
- Containment of radioactive substances (by means of the integrity of the pressure housing of the control rod drive mechanisms).

146 The proposed design is different to existing EDF and AREVA NPP designs in some parameters, by the design taking into account:

- Higher seismic loadings (Refs 26 and 27).
- Increased length to account for the enlargement of the active core.
- Detailed design improvements to aid EMIT.

147 In response to TQ-EPR-1461 (Ref. 10) and associated discussions, EDF and AREVA confirmed the CRDM design is in principle the same as that utilised in the German NPPs, which have approximately 30 years operational experience without encountering any major issues. The detailed design improvements have typically been of a minor nature, and have been integrated to address the variance in the design parameters stated above. Examples include the increase in the flange thickness, changes to the web detail between the housing, the operating coil assembly, displacement limiter design, and the collar design due to the change in the seismic loading. The collar design has been revised to position the housing weld in an area that allows for in service inspection. The drive rod, RCCA, pressure housing casing and the positional indicators have all increased in length to take into account the increase in the core length. Other improvements include the repositioning of electrical plug sockets, and the bolt design has also been changed.

148 I consider the information demonstrated and provided good evidence that the CRDMs are of an established design:

- Reliability is underpinned from operational experience and from carrying out research trials and the endurance test, with the summary results being captured in the EPR Short Drive Rod Configuration Synthesis Report (Ref. 28).
- Design improvements are typically of a minor nature.

149 My assessment of the technical specification (Ref. 29) has identified the design, material, fabrication, inspection and testing as a Q1 quality category (the highest category), and

these parameters are generally based on RCC-M (Ref. 51) and RCC-E (Ref. 52) codes and standards, which is aligned with my expectations against SAP ECS.3.

4.5.1.2 CRDM Design Change

150 In response to TQ-EPR-1462 (Ref. 10) and associated discussions, EDF and AREVA have explained design change (CMF-013, Ref. 48), which is associated with the Rod Cluster Control Assembly (RCCA). The RCCA comprises the control rods themselves, which are inserted into the Fuel Assemblies, and the RCCA spider, which attaches to the top of a set of control rods, and which connects to the CRDM drive rod.

151 The design change has been driven by a need to decrease the rod drop time in order to provide an increased margin in line with a change in the applicable design code, and to compensate for the reduction in length of the drive rod, due to plant spatial constraints. This necessitates an increase in the global mobile mass, with respect to the original mass, and the need to maintain or increase neutronic efficiency to ensure that any change is still bounded by RCCA Operational Experience Feedback.

152 The design solution adopted is to increase the length of the Silver-Indium-Cadmium (AIC) bar portion of the control rod, and reduce the length of the boron pellet section, to increase the mass; and compensate for the loss of neutron absorption by reducing the thickness of the boron pellet cladding (thereby increasing the actual boron pellet diameter, but keeping the outside diameter including cladding unchanged).

153 The revised design now incorporates AIC bar lengths and boron pellet designs which are standard for other plants worldwide, and so EDF and AREVA claim the design change is bounded by RCCA OEF.

154 As part of the design change EDF and AREVA have assessed the effect on the RCCA spider mechanical integrity, and also the Control Rod Drive Mechanism mechanical integrity, which they have found to remain within acceptable parameters. They also described OEF relating to the thinner boron pellet cladding, based on 3800 RCCAs, without detrimental reported effects.

155 The revised design has proved satisfactory from a seismic re-validation perspective, and the revised design has also proved satisfactory from a physical test perspective, covering rod vibratory behaviour, drag force, and endurance effects.

156 EDF and AREVA claim the design change to be acceptable, with the final analysis to be presented in the UK EPR RCCA design report.

157 I consider, that EDF and AREVA described a rational design change, accounting for Operational Experience Feedback, and I am satisfied that the design change can be incorporated within the UK EPR reference design for the GDA. However, I consider it to be an Assessment Finding (**AF-UKEPR-ME-07**) that the CMF-013 design change should be fully substantiated and reflected in all safety and design documentation.

4.5.1.3 CRDM Endurance Test

158 EDF and AREVA have provided information via TQ-EPR-1463 (Ref. 10) that covers a number of aspects on the CRDM endurance test plus the arguments that the RCCA design change (CMF-013, Ref. 48) is considered within the final test data.

159 Carrying out the test and early inspections found the latch tip to show excessive wear. As a consequence the latch tip material was changed to Stellite™ and the welding technique for attaching the tip to the latch changed from Tungsten Inert Gas (TIG) to an oxy-acetylene process. The welding process change is to aid the iron content to remain within the tip area, which results in the tip achieving increased hardness properties.

160 The information provides evidence that the drive rod, pressure housing, and coil housing remained within their design intent throughout the step endurance test. In addition the revised latch assembly remained within its design intent throughout its test.

161 The endurance test was stopped once 9 million steps were completed for the revised latch assembly design. The CRDM was still functioning within its design intent at the time.

162 Inspection identified that the stationary latches suffered single sided wear, while the movable latches suffered double sided wear.

163 The endurance test was carried out over the full length of the drive rod. I consider this does not accurately represent the operational duty of an NPP since during operations a specific length of the drive rod will be subjected to more contact with the latches. Therefore the drive rod may be subjected to a greater local wear than that reported within the endurance test. However, I recognise that during the operational phase, CRDMs are the subject of continuous condition monitoring and a malfunction (i.e. a missed step sequence) is immediately identifiable. Furthermore, there is no direct loss of safety function if a step is missed.

164 Inspection of the shims following the endurance test confirmed that they were in an acceptable condition from an integrity, geometric, and magnetic perspective. EDF and AREVA explained that the CRDM is of a design that ensures each shim has a clearance gap to its associated coil. This clearance gap mitigates impact loads on the shim during a stepping sequence.

165 My assessment has confirmed:

- The endurance test took into account plant operating environment parameters.
- Commercial consideration stopped the endurance test once the revised latch assembly had completed 9 million steps. The CRDM was still functioning within its design intent at the time.
- 300 drop tests have been carried out and each one successfully achieved its design drop time criteria.
- No stepping failure was experienced during the endurance test.
- Inspection of the revised latch unit subsequent to the 9 million steps showed 66% of the original nominal latch tip remained engaged.
- No latch tip broke off during the endurance test.
- No sub item was the subject of damage or failure that could impair the unit’s functional performance.
- There was no evidence of corrosion either during or following the test.
- The CRDM Endurance Test Report is progressing through its formal approval process.

166 The design intent for the proposed CRDM units to undertake 6 million steps is based on limiting EMIT requirements and the 60 year design life of the plant. This requirement is a significant increase when compared to the design intent of previous CRDM units. I am satisfied that the evidence collated in carrying out the endurance test is adequate to place a 6 million step claim on the CRDM units, with an adequate margin. It is important to state that the governing design life of the CRDM unit is 6 million steps, which EDF and AREVA claim will be adequate for 60 years. Although in reality there will likely be greater local wear on the drive rod than that represented by the test, EDF and AREVA stated that this was not considered the critical life limiting component within the assembly. Furthermore, the CRDM is subject to continuous monitoring, and there is no direct safety consequences associated with a missed step, which I accept as an appropriate argument.

167 My assessment has confirmed the endurance test took into account the design parameters associated with the RCCA Design change (CMF-013, Ref. 48) i.e. the increase in the mass of the RCCA.

168 EDF and AREVA also explained the CRDM seismic qualification test that:

- Verified that the drive rod can still drop at maximum static deformation, which is considered to be the bounding case from data derived from the seismic analysis.
- Demonstrated the latch unit is able to release the drive rod whilst distorted.
- Evaluated the control rod drop times for both normal and seismic scenarios.

169 In response to questions EDF and AREVA stated the positional indicator tests under a seismic event have not been conducted but are scheduled to be carried out. In the absence of this evidence, I consider this to be an Assessment Finding (**AF-UKEPR-ME-08**); evidence is required to be generated to show that the CRDMs meet their seismic design intent.

170 I also consider the absence of an approved copy of the CRDM Endurance Test Report that records the CRDM test evidence to be an Assessment Finding (**AF-UKEPR-ME-09**); approved CRDM Endurance Test Report to be generated.

171 However, notwithstanding these Assessment Findings, I am satisfied with the arguments and evidence provided by EDF and AREVA for this aspect in comparison to SAP EQU.1.

4.5.1.4 Safety Classification

172 The assignment of safety classification for the CRDM and the RCCA has been discussed at several technical meetings with EDF and AREVA.

173 My Step 3 assessment of the CRDM identified the latch assembly unit is an item that has an important role in terms of the CRDM being able to achieve its safety function.

174 The latch assembly 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. It is on a reactor trip that the latch assembly units are required to be repositioned to enable the RCCAs to drop and shut down the reactor.

175 My assessment of the PCSR has identified the latch unit not to be assigned with a safety classification, and it is my expectation that the latch mechanism is assigned with a safety classification of Class 1 against the UK SAPs.

176 The definition of a Safety Class 1 SSC is:

> "any structure, system or component that forms a principal means of fulfilling a Category A safety function."

177 In addition, my Step 3 assessment also identified the RCCA and drive rod as not being assigned with a safety classification, and it is my expectation that both items are assigned with a Safety Class 1.

178 In recognition of this shortfall against my expectations, two Regulatory Observations have been raised and issued. The Project has issued RO-UKEPR-043 (Ref. 11), Safety Categorisation and SSCs Classification, which is a cross-cutting RO and focuses on ensuring that an adequate methodology for safety categorisation and classification is applied across the whole of the UK EPR design. The second RO-UKEPR-056 (Ref. 11), CRDM Safety Classification, is a Mechanical Engineering specific RO, which is to ensure the output methodology of RO-UKEPR-043 is adequately implemented for each CRDM and RCCA.

179 Through discussion EDF and AREVA have accepted that both the latch unit assembly and the drive rod, plus the RCCA, are required to be assigned with a Safety Class 1. My assessment of the latest documentation (Ref. 32) provides the evidence that the latch unit assembly, the RCCA assembly, and the drive rod are assigned with a Safety Class 1, which now is aligned with my expectations against SAP ECS.2. However, the design code identification of "n.a." (assumed to mean "not applicable") to the drive rod and the displacement limiter is not to my expectations. I consider this to be an Assessment Finding (**AF-UKEPR-ME-10**); evidence is required that the CRDM and its constituent components are assigned with appropriate Mechanical Engineering design / material codes, which are commensurate to their importance to safety.

180 My assessment of the CRDM positional indicators confirmed that they are assigned with a Safety Class 1 (Ref. 32), which is aligned with my expectations.

4.5.2 Findings

AF-UKEPR-ME-07: *The licensee shall ensure that the RCCA CMF-013 design change is fully substantiated and reflected in all design and safety documentation. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-08: *The licensee shall generate evidence to demonstrate that the CRDMs meet their seismic design intent. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-09: *The licensee shall generate the approved copy of the CRDM Endurance Test Report that records the CRDM test evidence. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components -*

inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.

AF-UKEPR-ME-10: *The design code identification of "n.a." (assumed to mean "not applicable") to the CRDM drive rod and the displacement limiter is not to my expectations. The licensee shall generate evidence that the CRDM and its constituent components are assigned with appropriate Mechanical Engineering design / material codes, which are commensurate to their importance to safety. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

4.6 Isolation Valves Providing Containment Safety Function

181 During my Step 4 assessment, I further targeted the topic of isolation valves providing a containment safety function. This is due to their safety functions to provide adequate isolation and containment to control the spread of radioactive substances, and to open and close on demand to facilitate safe process operations and EMIT. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment Principle EDR.1 (Ref. 4) states 'Due account should be taken of the need for structures, systems and components important to safety to be designed to be inherently safe or to fail safe in a safe manner and potential failure modes should be identified, using a formal analysis where appropriate'.

182 I have reviewed the safety functions associated with these valves during my assessment, and the specific designs for their intended duty, to ensure that they are of a sound engineering principle, and have benefitted from appropriate Operational Experience Feedback.

183 My assessment of EDF and AREVA Rules for choice and codification of the valves (Ref. 33) and selection rules for valves and their actuator (Ref. 35) identified valves may be fitted with an electrical actuator (Motor Operated Valves). As a sample, I specifically selected to target EDF and AREVA's arguments and evidence that isolation valves fitted with electrical actuators fail in a safe manner, or adequate arrangements are incorporated into the design to manage the fault.

4.6.1 Assessment

184 The response to TQ-EPR-232 (Ref. 10) and discussions confirmed:

- Motor Operated Valves that have an F1 safety function (F1A or F1B) are connected to the diesel generator power supply.
- Motor Operated Valves that have an F2 safety function are considered for connection to the diesel generator power supply on a case by case basis and judged against the safety analysis with the decision generally depending on whether the valve is required to reach the final safe state after an RRC-A (Risk Reduction Category) event.

F1A, F1B, and F2 are architecture requirements associated with safety functional groups, in line with EDF and AREVA's safety function categorisation and SSC classification methodology, (Ref. 32).

185 In response to questions on a valve failing to close on demand due to a mechanical component failure within an actuator, rather than on loss of power, EDF and AREVA stated their design philosophy for isolation (containment safety function) is to have redundancy, by the system design incorporating two isolation valves in series. The SDM Extra Boration System (EBS) (Ref. 34) provides confirmatory evidence of this philosophy by a double isolation arrangement (e.g. valves 1350VB and 1410VB).

186 EDF and AREVA explained that electrical actuators are the preferred design choice over air operated actuators. Air operated actuators have been found difficult to qualify for a LOCA, plus the use of electrical actuators eliminates the management of exhaust air from the actuators within the containment.

187 They claim operational experience has identified globe valves as the preferred design choice over gate valves. It is also possible for a globe valve to be fitted with a bellows between the valve stem and body. The fitting of a bellows provides an additional containment barrier. The travel associated with the stem on a gate valve effectively prevents the incorporation of a bellows within a gate valve design.

188 In response to further questions, EDF and AREVA stated that when considering a change in a valve type, they carry out a design review, which gives due consideration to the French and German selection rules, and operational experience, prior to recommending a valve type for a particular application.

189 EDF and AREVA described the endurance testing specifications associated with valve designs, and in response to questioning explained that the number of cycles selected for the test is from experience that problems are likely to become apparent within this number of cycles.

190 My assessment of:

- Periodic test instructions of EBS (Ref. 40), which sets out the periodic tests performed on the EBS, which includes a number of valves, is generally aligned with my expectations from a GDA perspective.
- Completeness analysis of EBS periodic testing instructions (Ref. 41), which describes the concept periodic tests which support the EBS to ensure components achieve their safety functions, is generally aligned with my expectations from a GDA perspective.
- UK Classification of SSCs (Ref. 32) has confirmed the EBS branch pipes up to and including the first isolation valve are assigned with a Safety Categorisation A and a Safety Class 1. In addition the rest of the system has also been assigned with the same safety categorisation and classification, which is in line with my expectations.

191 In summary, EDF and AREVA have described a rational process, which has benefitted from Operational Experience Feedback. They have also provided evidence that valves have an adequate safety classification (Ref. 32), and I am satisfied that the assessed arguments and evidence meet my expectations against SAP EDR.1.

4.6.2 Findings

192 I have not identified any findings covering this area.

4.7 Check Valves

193 Check valves are of a passive design, (i.e. without the need for active initiation, operator intervention, or other support features (Ref. 4)), and are incorporated into many process systems with varying operational parameters. Consideration of the available Operational Experience Feedback is important to ensure the initial selection of valve type continues to be ALARP, and the design achieves its safety functional intent.

194 To progress my Step 4 assessment I particularly sought evidence of how EDF and AREVA have taken into account Operational Experience Feedback associated with check valves. This is in addition to reviewing the scope and frequency of EMIT and specifically inspections inside the valve body. I consider the following Safety Assessment Principles to be relevant to this aspect:

- Safety Assessment Principle EDR.1 (Ref. 4) states 'Due account should be taken of the need for structures, systems and components important to safety to be designed to be inherently safe or to fail safe in a safe manner and potential failure modes should be identified, using a formal analysis where appropriate'.
- Safety Assessment Principle EMT.1 (Ref. 4) states 'Safety requirements for in-service testing, inspection and other maintenance procedures and frequencies should be identified in the safety case.'

4.7.1 Assessment

4.7.1.1 Check Valves Operational Experience Feedback

195 My assessment of EDF and AREVA's Rules for choice and codification of valves (Ref. 33) identified that specific check valve types are the subject of design selection rules.

196 The response to TQ-EPR-233 (Ref. 10) provides EDF and AREVA's arguments to justify the rules and lists the valves that deviate from the rule with the appropriate justification. I consider the response to be rational and acceptable.

197 The response to TQ-EPR-611 (Ref. 10) and subsequent discussion provided details of EDF and AREVA's Operational Experience Feedback (OEF) associated with check valves. Examples of problems include design defects, fatigue not being adequately taken into account in the design, and materials not being suitable for high temperature applications. In addition OEF has identified that "dead leg" phenomena (a 'dead leg' is a section of pipework containing fluid which is not subject to normal process flow), introduces corrosion problems. The topic of "dead leg" phenomena is discussed in the section of this report relating to Good Engineering Practice.

198 The safety significant check valves for the primary circuit isolation are now based on swing check technology, with nominal bores ranging from 50mm to 300mm. The loadings on the check valves during operation are of thermal origin caused by thermal transients, and of mechanical origin caused by pressure and pipe loads etc. In particular, fatigue can be a significant phenomenon where there is significant thermal cycling. The valve design parameters, which are important in respect of fatigue are the valve body shape and the internal contours, which can lead to stress concentration effects.

199 The UK EPR check valve designs have an internal body with an increased fillet radius to overcome fatigue problems that were identified with the earlier check valve designs. Furthermore, the valve bodies for all check valves belonging to systems connected to the primary circuit have been changed from a cast, to a forging design. EDF and AREVA

claim this improves fracture toughness, reduces thermal ageing effects, and provides an increased resistance to fatigue. I have discussed this with the Structural Integrity assessment discipline and agree that increasing fillet radii will reduce susceptibility to fatigue effects, and that forging technology generally provides improved material properties.

200 In response to questions relating to the design process, EDF and AREVA stated that the fatigue loading / duty is initially defined by the process design discipline, and then passed through to the valve design organisation, who take on the responsibility to evolve the design to the design parameters. This is in line with my expectations.

201 In summary, I consider that EDF and AREVA have demonstrated a rational design improvement process based on Operational Experience Feedback, and I am satisfied with the improvements described from a GDA perspective against SAP EDR.1.

4.7.1.2 Check Valve EMIT Frequency

202 The response to TQ-EPR-611 (Ref. 10) explains the frequency of inspection of check valves in the main primary circuit and secondary systems. Inspections are required to be carried out at least once in every ten years. This is in accordance with current French regulatory requirements. However, operational experience has led to additional internal inspections being carried out every other outage on check valves that are positioned within dead leg pipework to manage and eliminate corrosion problems. In addition EDF and AREVA confirmed a visual external inspection is carried out at least once per refuelling outage, on valves that are within pressurised lines, with the external inspection focused on the area of the valve body and bonnet interface.

203 Internal inspection requires the removal of the valve bonnet to allow:

- Visual inspection of all sub components, moving components and the body internal surfaces.
- Dimensional checks to be carried out, if deemed necessary.
- Visual inspection for damage of the valve seating surface and stem guides.
- An investigation to be carried out for evidence of corrosion.
- The seal packing and any component that is out of specification to be replaced.

204 In summary, I consider that EDF and AREVA's inspection frequency of check valves and their depth of inspection are aligned with my expectations from a GDA perspective against SAP EMT.1. However, I expect check valve EMIT and frequency requirements to be documented as an integral aspect of the Assessment Finding related to this general topic area (**AF-UKEPR-ME-05**).

4.7.2 Findings

205 I have not identified any new Assessment Findings associated with this topic area.

4.8 Safety Relief Valves

206 I have reviewed the designs for the safety relief valves associated with the primary and secondary circuits associated with the UK EPR design. These valves are designed to

protect the circuits from overpressure events, and therefore contribute to the containment of radioactive substances safety function, as well as the heat transfer and removal safety function, by maintaining the structural integrity of the circuits. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment Principle EPS.3 (Ref. 4) states 'Adequate pressure relief systems should be provided for pressurised systems and provision should be made for periodic testing.'

207 I have also reviewed the low temperature over pressure protection design for the primary circuit due to the greater susceptibility of primary circuit materials to mechanical failure at low temperature. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment Principle EPS.4 (Ref. 4) states 'Overpressure protection should be consistent with any pressure-temperature limits of operation'.

4.8.1 Assessment

4.8.1.1 Safety Relief Valves

208 EDF and AREVA provided information for spring loaded safety relief valves in general, starting by explaining the design workflow process for valve design. Generally for valve designs they go to competitive tender, although for the pressuriser safety relief valve arrangement there is only one identified supplier. EDF and AREVA described the iterative nature of the design process, taking feedback from suppliers' technical parameters, and also interfacing with the safety team requirements. Specifically EDF and AREVA stated that the safety team are supplied with the design documentation to ensure that the safety requirements have been adequately captured.

4.8.1.1.1 Pressuriser Safety Relief Valve

209 The primary circuit pressuriser uses three Pressuriser Safety Relief Valves (PSRV). The PSRV design utilises two spring loaded pilots in parallel, one of which is normally isolated for maintenance, (plus two solenoid operated pilots in series). EDF and AREVA described the complex pilot valve operating design, and also discussed the testing campaign which had been undertaken in 2007, with no modification requirements identified.

210 In response to my questions EDF and AREVA stated that there are no recorded occurrences of spurious opening, failure to open, or failure to close, from operational experience for the type of PSRV used in the UK EPR design, based on OEF from this valve design installed in German NPPs.

211 I questioned EDF and AREVA as to why they had opted for a new spring loaded pilot detailed design for the UK EPR. They explained that OEF had suggested the requirement for this new pilot design in order to improve the leak tightness, and the choice had been made based on considerations of national preference plus the degree of qualification information which was available. I was satisfied with the explanation provided which I considered to be rational and pragmatic.

212 In response to my questions, EDF and AREVA stated that there was some Stellite™ used within the valve design seats. Alternatives had been considered, but they had

decided to retain Stellite™ for this application. EDF and AREVA stated that in their view Stellite™ was only a concern where the fluid flow was the subject of neutron flux in the reactor core; which was not the case for the Pressuriser Safety Relief Valve arrangement. It is true that these valve seats are in some contact with fluid (saturated steam) which is part of the primary circuit coolant, but I accept that significant contact is only made when the valve operates (which is infrequent), in which case fluid flow is out of the primary circuit. Given the safety importance of reliable operation of these valves, and the limited exposure of the primary circuit to Stellite™ in this application, I am satisfied that an adequate balance has been achieved.

213 In summary, I consider that EDF and AREVA have provided adequate information in this area and I am satisfied with the technical explanations provided. I have not identified any areas of concerns associated with the Pressuriser Safety Relief Valve design in comparison to SAP EPS.3.

4.8.1.1.2 Residual Heat Removal Spring Loaded Safety Relief Valve

214 EDF and AREVA also provided information regarding the Residual Heat Removal System spring loaded safety relief valves. They stated for the UK EPR the supplier had not yet been selected.

215 Specifically they described the tests which had been undertaken to validate the hydrodynamic arrangements to prevent valve chattering. EDF and AREVA stated that OEF from the French N4 NPPs with similar valves in the CVCS line is very good, with no instances of spurious opening, failure to open, or failure to close, reported.

216 I have not identified any issues of concern associated with this equipment, which represents a mature design and which has benefitted from appropriate Operational Experience Feedback, and am satisfied in comparison to SAP EPS.3.

4.8.1.1.3 Secondary Side Pressure Relief

217 EDF and AREVA also provided information on the secondary side steam relief arrangements, TQ-EPR-1442, (Ref. 10).

218 The secondary side steam relief arrangements comprise a 50% capacity Power Operated Relief Valve (PORV) arrangement of two valves in series, plus two separate 25% capacity spring loaded safety relief valves connected to the main steam line.

219 The information specifically focused on comparison with the German Konvoi design arrangement, and the French N4 arrangements. EDF and AREVA stated that for the Konvoi arrangement the main steam safety relief valve could be isolated upstream, which was considered unusual, but was as a result of local regulations. The Konvoi design also had a large number of pilots, and the overall valve configuration represented a complex configuration, (with multi-port arrangements).

220 EDF and AREVA stated that the EPR design had to be compatible with national regulations, and had been developed by detailed review of the existing Konvoi and N4 arrangements, to achieve the overall best solution. In response to my questions, they specifically stated that they only required 2 x 25% spring loaded safety relief valves since the pressure drops very rapidly and substantially in the event of a reactor trip (nominally within 1 second), and so this is used as a primary means of overpressure protection.

221 In the UK the Pressure Systems Safety Regulations (Ref. 30) are applicable in respect of pressure relief design requirements. In common with the theme of UK health and safety legislation these regulations are non prescriptive, and the relevant regulation states ‘The pressure system shall be provided with such protective devices as may be necessary for preventing danger; and any such device designed to release contents shall do so safely, so far as is practicable.’ – Regulation 4(5). The UK EPR does not provide secondary side 100% relief capacity via the two spring loaded safety relief valves, but EDF and AREVA do claim the reactor trip as the primary safety system, (in addition to the PORV and spring loaded safety relief valves). Furthermore, it should be recognised that the spurious operation of a secondary side safety relief valve is the initiating event for a fault sequence in its own right, leading to a cooldown fault (reactivity increase due to increased core moderation). I conclude that the secondary side over pressure protection system does meet these regulatory requirements, and that an appropriate balance has been achieved from a nuclear safety perspective.

222 EDF and AREVA have stated that these secondary side spring loaded safety relief valves for the UK EPR are derived from those installed on French NPPs. The design has benefitted from OEF through simplification, by removal of pneumatic opening/closing assist devices, which has been possible by adjustment of the valve set pressures. Since these changes were made on the French NPPs no occurrences of spurious opening, failure to open, or failure to close, have been reported.

223 In summary, I am satisfied with the design for the UK EPR secondary side steam relief arrangements against SAP EPS.3, and have not identified any areas of concerns for GDA.

4.8.1.2 Low Temperature Over Pressure Protection

224 I have also questioned EDF and AREVA in respect of the overpressure protection provided in the primary circuit at low temperature, due to the greater susceptibility of primary circuit material to mechanical failure at low temperature, TQ-EPR-1281 (Ref. 10).

225 This low temperature protection is provided by the Pressuriser Safety Relief Valves, mounted on the top of the pressuriser within the primary circuit. EDF and AREVA have also stated that the Residual Heat Removal System safety relief valves are not formally claimed as providing this safety function, although clearly they will have a benefit since their setpoint is below that of the PSRVs, leading to a conservative analysis.

226 The solenoid operated pilots within the PSRVs are actuated by the C&I system, based on the low temperature / pressure set points, to provide the pressure relief function. Only one PSRV (out of the three total) is required to limit the pressure for all transients, so that the arrangement is tolerant of a single failure, since at least two PSRVs must be available for appropriate operation.

227 EDF and AREVA have explained that the PSRVs must ensure the highest safety Category A, against the UK expectations, with an equipment classification of Class 1. This is in line with my expectations, and is reflected in the latest documentation (Ref. 32).

228 I am satisfied with the UK EPR design from a Mechanical Engineering perspective against SAP EPS.4, and have not identified any concerns associated with this requirement of low temperature overpressure protection.

4.8.2 Findings

229 I have not identified any findings covering this area.

4.9 Pumps

230 The UK EPR NPP contains numerous pumps within fluid transport systems, with associated safety functional requirements of varying degrees of importance. I have targeted my assessment on the Reactor Coolant Pump, and the Medium Head Safety Injection Pump, as being SSCs of particular safety significance.

4.9.1 Assessment

4.9.1.1 Reactor Coolant Pumps

231 I consider the role of the Reactor Coolant Pumps (RCP) to be important to safety. They are an integral part of the RCS containment barrier, and have the safety function of heat transfer and removal. I consider the following Safety Assessment Principles to be relevant to this aspect:

- Safety Assessment Principle ECS.2 states ‘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 regards to safety.’
- Safety Assessment Principle EQU.1 (Ref. 4) states ‘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.’
- Safety Assessment Principle EMT.1 (Ref. 4) states ‘Safety requirements for in-service testing, inspection and other maintenance procedures and frequencies should be identified in the safety case.’

232 During my Step 4 assessment I have continued to carry out assessment of the design of the Reactor Coolant Pumps, in particular targeting:

- The thermal barrier heat exchanger arrangement, due to its role in managing the temperature of the primary circuit coolant local to the pump seal system.
- The verification of the flywheel safety functional requirements, as the safety analysis requires the core to have sufficient coolant flow to avoid a Departure from Nucleate Boiling (DNB) event – under defined fault scenarios.
- The adequacy of ingress and egress provision for carrying out Examination, Maintenance, Inspection and Testing. I selected the replacement sequence of an RCP due to its size, mass and location within the plant; plus replacement of an RCP shaft seal system, which acts as an integral containment barrier to the reactor coolant primary circuit.

4.9.1.1.1 RCP Thermal Barrier System and Heat Exchanger

233 The response to TQ-EPR-1452 (Ref. 10) and associated discussions, clarified the thermal barrier heat exchanger design, which is integrated into the RCP shaft seal pump, and which ensures that the temperature within the shaft seal system is maintained at less than 95°C; (the RCP circulates reactor coolant water at ~ 300°C).

234 The thermal barrier heat exchanger comprises a set of circular tubes which are supplied with water from the Component Coolant Water System (CCWS). The RCP thermal barrier system also uses a set of thermal seals to limit heat transfer by conduction. In addition the seal injection flow for seal N°1 of the shaft seal system (which comprises three seals), also contributes to the required cooling function.

235 The thermal barrier system design is effectively identical to that used for the N4 series of reactors, although it now comprises a two stage heat exchanger to provide enhanced thermal performance.

236 The tests which have been undertaken on the OL3 NPP were classed as First Of A Kind (FOAK) for the RCP pump. These considered both the loss of seal injection water flow, and loss of CCWS flow to the thermal barrier heat exchanger. In both cases the tests provided satisfactory results, in line with the calculated predictions.

237 The alarms associated with the RCP thermal barrier system design are based on loss of seal injection and loss of CCWS flow to the thermal barrier heat exchanger. These comprise initial alarms, followed by activation of the Stand Still Seal System to maintain the containment barrier (if both the seal injection flow and CCWS flow are lost).

238 The thermal barrier heat exchanger design incorporates a double valve arrangement on both the inlet and outlet sides, which close in the event of tube rupture, to provide containment and thus prevent a LOCA from the primary circuit.

239 Provision of water to the thermal barrier heat exchanger from the overall CCWS system design architecture provides redundancy since the common auxiliaries are supplied from one train of the paired trains 1+2, and 3+4. Switchover from trains 1 to 2; or 3 to 4, is achieved by fast acting valves in order to limit the thermal transient.

240 In respect of the fault tolerance of the overall RCP cooling design, EDF and AREVA claim that the RCP design temperatures are not exceeded in the event of the following fault scenarios:

- Loss of Chemical and Volume Control System (CVCS) water injection associated with the N°1 seal system.
- Loss of CCWS flow to the thermal barrier heat exchanger.
- Loss of both of the above, if one function is recovered within two minutes.

241 The CVCS ensures continuous injection of cooled and purified water to the first stage of the reactor coolant pump seals and returns the leakage to the CVCS. In the case of a loss of the CVCS seal injection the thermal barrier heat exchanger cools the reactor coolant that flows up past the pump shaft prior to it coming into contact with the seal system.

242 Response to TQ-EPR-1243 (Ref. 10) confirms that on the loss of the thermal barrier heat exchanger, the FA3 operating technical specifications require the operator to:

- Shut down the reactor immediately to a normal shutdown state with the steam generators if the seal injection is not available.
- Reach a normal shutdown state on Residual Heat Removal (RHR) mode within 8 hours if the stand still seal system is not available.
- Reach a normal shutdown state on RHR mode within 3 days if the seal injection and the stand still seal system are available.

243 TQ-EPR-1243 (Ref.10) response also claims it is possible from a mechanical point of view to maintain an RCP in operation on the loss of the CVCS providing the associated thermal barrier heat exchanger is operable. I consider from a Mechanical Engineering perspective the CVCS seal injection is an important means of fulfilling the RCS containment function at the RCP seal and on its loss the pump and reactor should go into an appropriate shutdown state, (unless a suitably robust justification can be provided for continued operation). I consider this to be an Assessment Finding (**AF-UKEPR-ME-11**); a future licensee is required to clarify and justify the operating limits and conditions of the reactor and the RCPs on the loss of the CVCS system or the thermal barrier heat exchanger.

244 Assessment of EDF and AREVA proposed Classification of SSCs (Ref. 32) states the assignment of safety categorisation A and a safety classification of 2 to the thermal barrier heat exchanger lines and valves, which is in line with my expectations.

4.9.1.1.2 RCP Flywheel Design and Safety Functions

245 EDF and AREVA have explained that the UK EPR pump design has evolved from the N24 pump. This is fitted into the N4 NPPs with significant experience gained from the development of the tests carried out on the N24 pump, which at the time was treated as a First Of A Kind Pump. The units have been in operation for over fifteen years with no problems experienced associated with low flow characteristics, noting the system characteristics are verified during each outage prior to the plant going back to power.

246 The UK EPR RCP has an approximately 10% increase in flow characteristics when compared to the N4 NPP pumps. The supplier of the UK EPR RCP will be the same supplier as for the FA3 and OL3 NPPs, with the OL3 pump considered and treated as a First Of A Kind Pump.

247 The flywheel is located outside the main reactor coolant pressure boundary. It is of a sandwich construction of two steel plates and is attached and secured to the reactor coolant pump shaft by three mechanical keys.

248 The safety analysis requires the core to have sufficient coolant flow to avoid a Departure from Nucleate Boiling (DNB) event under defined fault scenarios. There is a requirement to manage unnecessary reactor trips due to short term grid oscillations and faults, and the design takes into account margins associated with:

- A loss of off-site power scenario (when the RCPs remain coupled to the switchboard, leading to a worst case flow reduction).
- Uncertainties on the RCP characteristics and primary circuit head losses.
- Uncertainties on the simulation of the RCP behaviour within the computer code.

249 The flywheel is claimed to be a High Integrity Component by EDF and AREVA; (this aspect of the claim is outside the remit of the Mechanical Engineering assessment discipline). The design incorporates six through holes, located at positions which would facilitate inspection of the most highly stressed areas, (which are the keyway corners), although the EMIT regime is yet to be confirmed. This assessment aspect is being led by the Structural Integrity discipline.

250 Building on the experience of the N4 pumps the UK EPR inertia requirements have been determined to be a minimum of 5210 Kg.m^2. The RCP inertia is taken from the sum of

the motor flywheel, the motor rotor and the pump rotor, and also takes into account manufacturing tolerances. EDF and AREVA state that the design requirement is cascaded through to the supplier via the System Design Manual and the equipment specification.

251 Response to TQ-EPR-1180 (Ref.10) confirms the assignment of a Safety Classification 1 to the flywheel, which is in line with my expectations against SAP ECS.2, and which is now reflected in the latest safety categorisation / classification documentation (Ref. 32).

4.9.1.1.3 RCP Flywheel Verification

252 The response to TQ-EPR-1180 (Ref. 10) and associated discussions have clarified:

- The process followed to determine and specify the primary coolant heat transfer safety functional requirements.
- The process followed to substantiate the flywheel safety functional requirements through the design process i.e. the concept design phase (empirical and / or theoretical), full size test loop pump qualification, site commissioning etc.
- The site flow coast-down test.
- N4 NPP feedback of operational experience.

253 Verification of the flywheel design is carried out at the detailed design stage by mechanical analysis that gives consideration to stress loadings, brittle fracture and the connection detail between the flywheel and the motor shaft.

254 The mechanical analysis is carried out against two conditions, the normal operational speed and a 25% overspeed condition.

255 EDF and AREVA confirmed the factory acceptance loop test is carried out at normal plant operational pressure and temperature, but the test is limited to the:

- Functional testing of the shaft seal system.
- Operation of the Stand Still Seal System.
- Confirmation that the pump has been assembled correctly and is fully functional. The test is carried out with a reduced flow of $50m^3$/hr, noting that NPP operational flow is circa 28 000 m^3/hr. The test is undertaken for a nominal 30-40 hrs, although the first pump for OL3 NPP was tested for nominally 250 hrs (based on being classed as a First Of A Kind pump).

256 During commissioning a system test is carried out at normal operational parameters and includes a:

- Full flow test.
- Shaft seal test.
- Stand Still Seal System test.
- Flywheel coast down test.

In addition the flywheel coast down test is carried out for two conditions:

- Test condition 1 - Main line electrical breaker open.

i) The RCPs remain coupled to the switchboard, leading to a faster flow reduction due to electrical coupling effects.

ii) Similar to a Loss of Offsite Power (LOOP) and is consistent with the safety analysis.

iii) Negative impacts on the other systems (e.g. a loss of vacuum at the condenser).

- Test condition 2 - Individual RCP electrical breakers open.

i) The 4 RCPs are disconnected from the switchboard simultaneously, which leads to a slower flow reduction. The slower flow reduction has to be calibrated for this scenario, since this is the test undertaken during outages, and the more onerous LOOP scenario has to be verified.

ii) No impact on other systems.

257 EDF and AREVA claim the correction factor between the two tests is in the order of 2.5%. Once an NPP is operational, experience has led to test condition 2 being undertaken as the preferred test, and applying the correction factor appropriately to the results. During each outage the flywheel test is carried out with the individual RCPs disconnected from the switchboard. The resulting parameters are then adjusted to take into account the electrical coupling effects, and the results are then compared to the safety analysis.

258 Once an NPP is operational, verification of the flow decrease behaviour is carried out during each outage by carrying out a test in accordance with the following criteria, which is a similar approach to that utilised on the existing French fleet of NPPs:

- Reactor is in a hot shutdown mode.
- Verification is performed on the sizing transient in terms of DNB and LOOP.

259 EDF and AREVA explained that the system does not offer any direct flow measuring facilities; the RCP speed sensors are utilised. EDF and AREVA claim the decrease in speed from the RCP sensors can be correlated to the decrease of flow, which I consider to be reasonable.

260 They also described the RCP factory tests in Gennevilliers for the N24 pumps. The aim of the test was to collate and understand the coast down time of an RCP with and without operation of the oil lift pump. The oil lift system function is to provide a lubrication barrier between the bearing mating surfaces to ensure good operational characteristics and to preserve their design life. EDF and AREVA claim the test results were in-line with their expectations.

261 As detailed design, FATs and SATs are considered to be outside the scope GDA and part of Phase 2, I consider this aspect falls under the identified generic Assessment Finding on the topic (**AF-UKEPR-ME-01**).

262 In summary, notwithstanding the Assessment Findings, I am satisfied with the design and testing methodology associated with the RCP flywheel against SAPs EQU.1 and EMT.1, including due consideration to operational experience, from a Mechanical Engineering perspective.

4.9.1.1.4 RCP and Seal Replacement

263 EDF and AREVA have explained the sequences involved in replacing the reactor coolant pump motor that has a mass in the order of 60 tonnes (65 tonnes lift inclusive of the lifting beam), and the reactor coolant pump seals.

264 The information clarified the design improvements that have been incorporated from the experience gained from the OL3 project. Such design improvements include.

- The re-routing of ventilation ductwork to eliminate the requirement to remove a ventilation spool section.
- Platform re-design to minimise the number of sections that are required to be removed to carry out the maintenance, i.e. handling improvements.

265 In response to questions, I noted the pump supplier recommends the electric motor is refurbished every 12-15 years. With the design incorporating four pumps and assuming an 18 month outage regime, this constraint is likely to introduce the requirement to start replacing a motor between the fifth and the seventh scheduled outage and for the subsequent three scheduled outages. Then there will be a further period of circa four outages when the motors are expected to be within their design intent. Then the motor replacement sequence is reintroduced. This pattern is then continued throughout the operational phase of the plant, which I consider may lead to the requirement to replace circa twenty pump motors over an NPP design life of sixty years, which I consider is a significant requirement.

266 My assessment has identified that the design does not specifically consider a drop load scenario as being a credible event for the RCP replacement. EDF and AREVA stated that their design strategy is for the polar crane to be of a sufficiently high integrity design that a drop load is not credible. UK regulatory experience indicates that significant events are often associated with rigging faults, or load path faults. It is my expectation that EDF and AREVA should systematically assess the possibility of rigging faults and load path faults for lifts of nuclear safety significance, and identify and implement reasonably practicable improvements to their design. This review should also include identification of equipment vulnerable to load interaction, and associated identification of reasonably practicable measures to either eliminate this hazard by design, or protect equipment as appropriate. This should be conducted against the background of UK Operational Experience Feedback in this area. This aspect is discussed further under the Crane topic area later in this report as the topic forms an integral aspect of RO-UKEPR-052 (Ref. 11) – Nuclear Lifting Rigging and Load Path Faults and RO-UKEPR-070 (Ref. 11) – Internal Hazards and Dropped Loads.

267 My assessment has not identified adequate evidence to demonstrate that RCP maintenance activities meet applicable Conventional Safety Regulations. During the pump seal replacement activities several sequences involve the manoeuvring of plant and equipment that exceed manual handling lifting limits. EDF and AREVA's explanation did not specifically cover the conventional safety regulations and requirements which are relevant for this type of activity in the UK. I consider conventional safety regulations are pertinent to this activity and EDF and AREVA are responsible for ensuring the design achieves the applicable regulations, e.g. The Construction (Design and Management) Regulations 2007 (CDM), the Lifting Operations and Lifting Equipment Regulations 1998 (LOLER), the Provision and Use of Work Equipment Regulations 1998 (PUWER) etc). I consider this to be an Assessment Finding (**AF-UKEPR-ME-12**); a future licensee to

provide evidence to demonstrate the RCP maintenance activities meet applicable Conventional Safety Regulations.

268 EDF and AREVA also claim that PCSR Chapter 18 identifies which aspects of the pump require routine maintenance; discussion confirmed that the impellor and the shaft have a 60 year design life, which is substantiated with operational experience of existing plants, technical analysis and aging calculations.

269 Following assessment of Chapter 18, I consider there to be limited detail to understand the pump's maintenance requirements. I consider this to be an Assessment Finding (**AF-UKEPR-ME-13**), although I note that the topic is also an integral part of cross-cutting Regulatory Observation RO-UKEPR-055, (Ref. 11) Plant operating limits and maintenance schedules.

4.9.1.2 Medium Head Safety Injection Pump

270 Following consultation with my Fault Studies colleagues, I have also targeted assessment of the Medium Head Safety Injection (MHSI) pump due to its safety significance in reactor core residual heat removal, reactivity control, and containment for a number of fault events. I consider the following Safety Assessment Principles to be relevant to this aspect:

- Safety Assessment Principle EQU.1 (Ref. 4) states '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.'
- Safety Assessment Principle ECS.1 (Ref. 4) states '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'.
- Safety Assessment Principle EMT.1 (Ref. 4) states 'Safety requirements for in-service testing, inspection and other maintenance procedures and frequencies should be identified in the safety case.'

271 My assessment of the Medium Head Safety Injection pump targeted the following aspects:

- Role and safety functions.
- Pump sizing methodology and qualification.
- Examination, maintenance, inspection and testing regime.
- Consideration to Operational Experience Feedback.

4.9.1.2.1 MHSI Pump Design and Safety Functions

272 Response to TQ-EPR-1445 (Ref. 10) provides information on the role and safety functions associated with the Medium Head Safety Injection (MHSI) pumps.

273 The MHSI pump has the following safety functions:

- Core and containment residual heat removal.
    i) To limit the draining of the RCS in the event of a specified design basis Loss Of Coolant Accident (LOCA). This is required to avoid boiling of the reactor coolant.

  ii) To supply cold water to the RCS in the event of a specified design basis LOCA. This is to manage core uncovering following a break and to provide adequate coolant to remove heat from the core.

  iii) To feed and bleed the RCS with the dedicated pressuriser lines in Risk Reduction Categories RRC-A and RRC-B events. This is required when the RHRS is unavailable.

- Core reactivity control.

  i) Prevent Steam Generator Tube Rupture (SGTR) reverse flow from SG secondary side to the RCS. This is to avoid and limit unacceptable RCS dilution.

  ii) Prevent departure from nucleate boiling in the event of a steam line break by limiting the RCS pressure depletion or level compatible with non departure from nucleate boiling. This is to ensure the heat is transferred to the fluid and there is no risk of fuel failure.

- Containment of radioactive substances.

  i) Provide sufficient injection flow in the event of a design basis LOCA. This is required to avoid containment design pressure or in containment qualification temperature limit being exceeded.

  ii) Not challenge the Main Steam Supply Valve (MSSV) in the event of a design basis SGTR. This is to prevent containment bypass risk.

274 The MHSI pump has the following operational functions:

- Operational Functions - State A - Normal power operations.

  i) Filling the accumulators to the required level from the In-containment Refuelling Water Storage Tank (IRWST).

- Cold Shutdown - State E, core unloaded.

  i) Filling the reactor pit prior to reloading.

  ii) Filling the accumulators after periodic testing from the IRWST.

275 My assessment considers that, from a Mechanical Engineering perspective, appropriate consideration has been given to:

- Single failure criterion; the whole system is split into 4 individual trains, 1 per loop and system isolation is provided.
- Physical separation; each train is located in a different building compartment.
- Emergency power supply is provided from the main diesel generators.
- Periodic testing is to be carried out on a two monthly basis and requirements are to be identified and cascaded into the plant EMIT requirements.
- Seismic; the system is SC1 Seismic classified.
- System sizing methodology; fault analysis set concept bounding criteria with the pump supplier considering the available pump technology to achieve the design criteria during the detailed design phase.

276 Responses to questions clarified that the system design parameters are similar to existing systems across the range of the French Fleet of operating NPPs.

277 Examples of such parameters include the system design pressure of up to 105 bar, temperatures of up to 120°C, and various operating flow rates and load cases for both thermal and pressure transients, and stress levels.

278 My assessment and associated discussion confirmed the following technical matters:

- The pump design parameters ensure sufficient Net Positive Suction Head during a design basis event.
- The system incorporates a mini flow bypass to ensure a sufficient flow rate is available for effective pump operations.
- Analysis has been carried out to understand the pumps limitations due to the process fluid containing debris.
- During a State A power operation, control of core reactivity is ensured against design basis LOCA events if the MHSI pump system delivers greater than $92m^3/h$ at a pressure of 60 bar abs.
- During States C, D and E the pump mini flow bypass arrangement manages the system lower pressure requirements.
- The system design has evolved around the system performance criteria, the available pump technology and the system design validation process.
- The pump technology is selected by the supplier under the control of EDF and AREVA, and is dependant on the confidence in achieving the performance criteria.

279 In response to questions EDF and AREVA confirmed the following:

- The Flamanville 3 and the Olkiluoto NPP pumps are of the same design and are being manufactured by the same supplier. The UK EPR pump design parameters will be identical, and the pump will be sourced from the same supplier.
- The pump detailed design is complete (for FA3 and OL3).
- The pump has been subjected to performance tests.
- The system design incorporates a suitable margin in respect of the Net Positive Suction Head (NPSH) requirements.
- The supplier selected a multistage centrifugal pump with a constant speed, fitted with an inducer. This pump matches performance criteria of:
  i) operating at 100 bar;
  ii) the requirement to increase flow rate when the system pressure is reducing;
  iii) the available NPSH at maximum flow rate and the pump safety significant reliability requirements.
- The pump is positioned horizontally, as Operational Experience Feedback indicates a horizontal pump is less susceptible to vibration.

4.9.1.2.2 MHSI Pump Validation

280 The system design is validated using a hydraulic design model, where the actual pump curve design information is fed back into the model to validate the system performance. This is in addition to a test and qualification program.

281 In response to questions, EDF and AREVA stated the following :

- the hydraulic model is utilised to analysis and check:
  - i) Supplier’s pump characteristics.
  - ii) System pressure losses.
  - iii) NPSH margins.
  - iv) Safety performance requirements.
  - v) Flow rate adjustment devices.
  - vi) Detailed technical parameters compared to the original concept values.
- the qualification program includes carrying out:
  - i) Concept hydraulic performance test to analyse the pump curve and mechanical power parameters.
  - ii) Thermal shock test to analyse the mechanical power, vibration effects, bearing temperatures, and seal leak rates. Following testing the equipment is disassembled and inspected.
  - iii) Detailed hydraulic performance tests to assess the pump curve and mechanical power.
  - iv) Endurance load (water and particle) test is carried out for 400 hours (for FOAK pump) to analyse mechanical power, vibration effects, bearing temperatures and seal leak rates. Following testing the equipment is dismantled and inspected.
  - v) Final confirmatory hydraulic performance test to assess the pump curve and mechanical power.
- seismic qualification is carried out by calculation and analysis.

282 The response to TQ-EPR-995 (Ref. 10) and discussion provided further information covering the qualification tests for the Medium Head Safety Injection pumps. These tests comprise hydrostatic tests, performance acceptance tests, and endurance tests; with the seals qualified by analogy to the Low Head Safety Injection (LHSI) pumps, which have a more onerous duty in this respect. These seal tests comprise thermoshock tests, endurance tests with hot water, and tests with particulate in water.

283 In response to questions, EDF and AREVA stated that the supplier for the MHSI pump and LHSI is the same, which is part of the consideration that the LHSI tests bound the MHSI requirements for the seals.

284 As a general statement, EDF and AREVA commented that equipment qualification is achieved by physical tests, Operational Experience Feedback (OEF), calculation, and / or by analogy. For new items, First of a Kind (FOAK) tests are undertaken, and a technical file is developed. This is then used as a reference point for determining if such tests require to be repeated in the event of changes etc, or whether verification by analogy is appropriate. Because of the size and cost of the pump, the equipment used for the

FOAK test is also supplied for operational use, subject to any minor refurbishment as necessary. The integrity of any refurbishment is assured through site commissioning tests. EDF and AREVA explained as a point of principle that the FOAK tests are to test the design, whereas subsequent Factory Acceptance Tests are to test that manufacturing and material quality are being achieved.

285 They explained that the test procedure / requirements are developed initially by the designers, which are then passed to the pump supplier. The procedure is then verified / validated via an iterative feedback loop involving all three parties.

286 EDF and AREVA described in detail the:

- Hydrostatic test, which they claim is a routine test applied to all production pumps. The acceptance criteria are: no permanent strain and no leakage after 30 minutes, and a conformance test certificate is supplied as evidence of acceptability.
- Performance tests, which cover:
  - i) head vs flow rate;
  - ii) power vs flow rate;
  - iii) efficiency vs flow rate;
  - iv) Net Positive Suction Head (required) vs flow rate;
  - v) bearing vibration;
  - vi) bearing temperature.

287 They then described the supplier's specialised factory test loop used for the performance acceptance tests, noting that under plant EMIT requirements, a different series of tests will be performed due to the limitation inherent within the plant design.

288 The endurance test comprises of the following:

- A 400 hour test for the FOAK pump with 50 starts / stops and 50 flow variations.
- Other pumps 20 hour test with 10 starts / stops and 10 flow variations.

289 At the end of the test, the pump is fully dismantled for inspection, including a visual check of parts and dimensional check of clearances. The following parameters are measured during the test:

- bearing temperatures;
- fluid temperatures (and ambient), and flow;
- motor power;
- speed;
- bearing vibrations;
- motor winding temperatures;
- mechanical seal performance.

290 In response to questions, EDF and AREVA stated that the 400 hour test duration selection was based on test experience to date, where design related problems if apparent, manifested themselves between 200 ~ 300 hours.

291 EDF and AREVA provided technical detail for the LHSI seal tests, which includes a 240 hour endurance test, various flow rates and temperature steps, and a 400 hour test with water borne particulate including for back flushing. Following these tests the pump is dismantled and inspected.

292 As FATs and SATs information are considered to be outside the scope GDA and part of Phase 2, I consider this aspect for the MHSI pump falls under the identified general Assessment Finding on the topic, (**AF-UKEPR-ME-01**).

293 The response to TQ-EPR-996 (Ref. 10) describes the condition monitoring that is fitted to the system. EDF and AREVA confirmed that during the NPP lifetime periodic testing of the pump's performance head vs flow rate will be carried out. No specific equipment is required to be fitted to the pump, as the sensors used are the ones already incorporated within the safety injection system i.e. flow meters and pressure measurements across the orifices. In addition while carrying out a periodic test other parameters are monitored to confirm the pump is operating within its design intent. To support these periodic tests the pump design incorporates:

- Measuring nipples for fitting vibration sensors onto the pump and motor bearing housing.
- Pump bearing temperature sensors.
- Motor bearing temperature sensors.
- Stator winding temperature sensors.

294 In response to questions, EDF and AREVA described several examples of preventative maintenance activities. I acknowledged that this is an area where work is in progress, but I was provided with a satisfactory level of confidence that a process is in place to identify the aspects that support the system reliability and availability.

295 An example of periodic testing included:

- The RIS-FS-G Safety function (MHSI RCS cold leg injection with the mini flow line open) which will test the:
  - i) MHSI pump actuation and associated delay in the pump to start.
  - ii) Valves and check valves for operation and the injection flow rate.
  - iii) Sensors and associated alarm signals from the equipment being tested.

296 In summary, notwithstanding the Assessment Findings, I consider that EDF and AREVA provided a good description of an acceptable and thorough test process, and design principle and specification for the MHSI pump, and I am satisfied in this technical area from a GDA and a Mechanical Engineering perspective against SAPs EQU.1 and EMT.1.

4.9.2 Findings

AF-UKEPR-ME-11: *The licensee shall clarify and justify the operating limits and conditions of the Reactor and the Reactor Coolant Pumps on the loss of the Chemical and Volume Control System seal injection system and / or the thermal barrier heat exchanger. Target Milestone – fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-12: *The licensee shall provide evidence to demonstrate the Reactor Coolant Pump maintenance activities meet applicable Conventional Safety Regulations. Target Milestone – fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-13: *The licensee shall ensure that the Reactor Coolant Pump maintenance requirements are adequately specified to meet the safety functional requirements throughout their operational life. Target Milestone – fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

4.10 Cranes

297 Lifting of nuclear packages or lifting operations over nuclear safety significant plant and equipment is an intrinsically hazardous, yet necessary activity within a Nuclear Power Plant, and I have continued my assessment in this important area. I consider the following Safety Assessment Principles to be relevant to this aspect:

- Safety Assessment Principle ECS.3 (Ref. 4) states '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.'
- Safety Assessment principle EDR.1 (Ref. 4) states 'Due account should be taken of the need for structures, systems and components important to safety to be designed to be inherently safe or to fail in a safe manner and potential failure modes should be identified, using a formal analysis where appropriate.'
- Safety Assessment Principle EDR.2 (Ref. 4) states 'Redundancy, diversity and segregation should be incorporated as appropriate within the designs of structures, systems and components important to safety.'

298 Specifically I have focused my attention on the following four lifting systems, based on my consideration of their high safety importance:

- The main containment Polar Crane.
- The spent fuel pool area Auxiliary Crane.
- The main containment Re-fuelling Machine.
- The spent fuel pool area Spent Fuel Mast Bridge.

It should be noted that for the UK EPR design, the transfer of spent fuel out of the Nuclear Power Plant is undertaken using a dedicated low level transfer facility, which is covered in Section 4.15 this document, and as such does not present the associated hazard of heavy lifting of a spent fuel cask.

4.10.1 Assessment

4.10.1.1 Mechanical Design Features

299 I have undertaken a limited assessment of the Mechanical Engineering design of the four crane systems in terms of power transmission arrangements, reeving systems, and

associated design safety factors, TQ-EPR-921, TQ-EPR-1301, TQ-EPR-1357 (Ref. 10). I consider the design of lifting systems for nuclear application to be a relatively mature area of engineering, and recognise that EDF and AREVA have considerable design and operating experience in this field.

300 For the Polar Crane the transmission arrangements for the main 320 tonne hoist comprise a ‘closed loop’ drive train, comprising the drive motor, and two reduction gearboxes connecting to the main drum equipment for winding the hoist rope. This design uses one operational brake and one emergency brake for each gearbox. The secondary hoist (35 tonne) and the auxiliary hoist (5 tonne) both comprise an ‘open loop’ drive system. Both these hoists have an operational brake and a secure motorised movement system connected to the main drum.

301 For the Polar Crane main and secondary hoists, a cross reeving system is utilised, which uses two separate rope systems to ensure that the load is lifted in a purely vertical sense as the ropes wind on the main drum; and is also specifically designed to ensure that a symmetrical loading on the hoist lower block is maintained in the unlikely event of rupture of one of the two rope systems. This is in line with my expectations for such heavy load handling equipment. For the auxiliary hoist a dual reeving system is used, which uses two rope systems and ensures a purely vertical lift, but which does not provide the symmetrical load pattern on the hoist block in the event of rupture of one of the ropes, (noting that the lower block only uses two sheaves (pulleys)). For each hoist the fixed point is equipped with dampers to absorb any shock associated with a rope rupture. In the event of a rope break, an alarm is displayed and the hoisting movement is stopped. The hoisting operation can then be restarted to move to a safe state, but only utilising lowering movement.

302 For the Spent Fuel Pool (SFP) Auxiliary Crane the transmission arrangements utilise an ‘open loop’ system comprising a motor, a single gearbox, the winding drum, plus an operational and auxiliary brake on the motor side of the gearbox, and a safety brake on the drum. A power train monitoring system is also used to sense the loss of synchronisation between high speed motor shaft, and the low speed drum shaft.

303 For the SFP Auxiliary Crane, a dual reeving system is used and the fixed point is equipped with dampers to absorb any shock associated with a rope rupture. In the event of a rope break, an alarm is displayed and the hoisting movement is stopped. The hoisting operation can then be restarted to move to a safe state, but only utilising lowering movement.

304 For the Re-fuelling machine and Spent Fuel Mast Bridge a dual reeving system is also used, and in the unlikely event of a rope breakage then the operation can be completed utilising only the remaining rope, but this requires the use of a dedicated bypass.

305 EDF and AREVA have stated that the Polar Crane, the Auxiliary Crane, the Re-fuelling machine and the Spent Fuel Mast Bridge will be designed to either the German KTA design code (Ref. 44), or the French BTS / BTR codes. I have questioned EDF and AREVA on the safety factors inherent within these codes and in particular on the residual safety factor in the remaining rope in the event of failure / rupture of one of the rope systems. I am satisfied with the responses and explanations which have been provided.

306 I am satisfied with the Mechanical Engineering design features of the UK EPR lifting systems against SAPs ECS.3 and EDR.2

4.10.1.2 Rigging and Load Path Faults

307 I consider that the likelihood of mechanical failure due to inherent defects within the lifting systems to be very low, due to the rigorous quality assurance regimes to be applied during manufacture, and associated level of EMIT applied during the lifetime of the plant, including test lifts as appropriate. In this respect it should be noted that the Polar Crane is used extensively during the initial construction phase for installation of the Reactor Pressure Vessel, the Steam Generators and Pressuriser etc, and so any significant issues would be identified at this stage when there is no nuclear hazard. Operational Experience Feedback from the UK and also the IAEA Incident Reporting System (IRS) indicates that the great majority of nuclear lifting abnormal events are associated with operational errors. Initial discussions with EDF and AREVA identified that this area was not adequately recognised or justified to my expectations, and I have therefore focused the majority of my assessment on the area of rigging and load path / route faults (i.e. the route taken by the load during the lifting / lowering / translating operation). In pursuing this line of enquiry I have also coordinated with the Internal Hazards assessment discipline, which has raised similar concerns, specifically associated with the overall design consideration of dropped loads within the NPP. The Internal Hazards discipline has raised a Regulatory Observation, RO-UKEPR-070 Dropped Loads and Impacts (Ref. 11), associated with this area, and has sought advice from the Nuclear Directorate's Mechanical Engineering Topic Group as part of the progression of this Regulatory Observation line of enquiry. In particular the topic group provided the following assessment advice:

- Crane and lifting equipment reliability is determined by many factors in addition to equipment integrity. Regardless of integrity claims it is considered necessary to assess the consequences of dropped loads and other malfunctions.
- The operating limits and conditions for cranes and lifting equipment should be determined taking account of the failure consequences assessment, and industry and regulatory guidance and engineering good practice, and operation should be demonstrated to be ALARP.

This advice is in line with my judgement, and I have supported the Internal Hazards discipline in this area, which is reported in the relevant Internal Hazards Step 4 Report as appropriate.

308 I have raised a Regulatory Observation, RO-UKEPR-052 (Ref. 11), Nuclear Lifting Rigging and Load Path Faults, to require EDF and AREVA to provide an adequate justification covering these aspects. This Regulatory Observation was raised in February 2010, and is summarised as follows:

- Assessment to date of the nuclear lifting arrangements and associated mechanical design philosophy has identified an apparent lack of systematic review of rigging and/or load path faults, to either preclude them by design, or minimise their frequency by the use of mechanical equipment. This review should also include identification of equipment vulnerable to load interaction, and associated identification of reasonably practicable measures to either eliminate this hazard by design, or protect equipment as appropriate.

309 Three Regulatory Observation actions were raised associated with this RO, described as follows:

- EDF and AREVA to systematically review the rigging arrangements for all lifting equipment associated with lifts of nuclear safety significance, to identify faults, and review and implement reasonably practicable improvements to either eliminate such faults by design, or limit their frequency by the provision of engineered protection systems.
- EDF and AREVA to systematically review the load path for all lifts of nuclear safety significance, to identify the potential for load interference (e.g. snagging or ledging), and review and implement reasonably practicable improvements to either eliminate such faults by design, or reduce their frequency. This review should also identify equipment vulnerable to load interaction, and review and implement reasonably practicable improvements to either remove this hazard by design, or reduce the consequence by appropriate protection measures.
- EDF and AREVA to review Operational Experience Feedback associated with UK nuclear lifting operations, and identify and implement any reasonably practicable improvements to their design.

310 EDF and AREVA have provided a response to this Regulatory Observation, initially through document 'Review of OEF associated with UK nuclear lifting operations', (Ref. 21). EDF and AREVA have concluded that most of the issues identified were associated with operator error or misuse of the equipment, and did not identify any specific engineering improvements for their lifting systems on the basis of this study. I accept this conclusion.

311 EDF and AREVA have responded to the first two actions through document 'UK EPR GDA – Management of Nuclear Safety Significant Lifting', (Ref. 22). This document has analysed the rigging and load path / route concerns for the following four cranes, in line with discussion held during technical meetings:

- The main containment Polar Crane.
- The spent fuel pool area Auxiliary Crane.
- The main containment Re-fuelling Machine.
- The spent fuel pool area Spent Fuel Mast Bridge.

312 For the Polar Crane, based on considerations of hazard (i.e. consequence), EDF and AREVA have considered the following nuclear safety significant lifts:

- Lifting of reactor cavity cover slabs.
- Lifting of reactor building pool stop gate.
- Lifting of Multiple Stud Tensioning Machine.
- Lifting and transport of the Reactor Pressure Vessel closure head.
- Lifting and transport of the RPV lower and upper internals.

For the Auxiliary Crane, based on considerations of hazard, EDF and AREVA have considered the following nuclear safety significant lifts:

- Handling of the Fuel Pool Stop Gate (penstock) in the spent fuel pool area.
- Handling of spent fuel assemblies (backup to the Spent Fuel Mast Bridge).

For the Re-fuelling Machine and the Spent Fuel Mast Bridge, EDF and AREVA have considered handling operations associated with spent fuel assemblies. The scope of the study is in line with my expectations.

313 In terms of the rigging arrangements for all the lifts identified above, EDF and AREVA have analysed the equipment designed to date, and have identified, described and discussed features to ensure integrity of the load attachment, including 'poka yoke' features which help prevent inadvertent assembly of the rigging to load interface. EDF and AREVA have stated that some of the rigging equipment detailed design is not complete at this stage of GDA, and I consider it to be an Assessment Finding (**AF-UKEPR-ME-14**) that this should be completed with due regard to this Regulatory Observation. Notwithstanding this Assessment Finding, I am satisfied with the response to this aspect of the Regulatory Observation, as provided by EDF and AREVA.

314 In terms of load path (load route) considerations, I have reviewed the EDF and AREVA response and have identified three nuclear lifts where there are significant ALARP options in terms of the load path chosen. These lifts are:

- Lifting of the RPV missile protection slabs by the Polar Crane.
- Lifting of the RPV closure head by the Polar Crane.
- Lifting of the Fuel Pool Stop Gate by the Auxiliary Crane.

315 In respect of lifting of the RPV missile protection slabs, EDF and AREVA have identified two possible load paths: Case 1 which considers the main vertical lift to be undertaken away from the RPV area; Case 2 which considers the main vertical lift to be undertaken above the Reactor Building Pool, using the integrity of the remaining slabs to provide protection in the event of a dropped load. EDF and AREVA have stated that the final design selection of load path will be undertaken following conclusion of dedicated studies regarding dropped load consequences, which are being undertaken in response to RO-UKEPR-070. I consider it to be an Assessment Finding (**AF-UKEPR-ME-15**) that this exercise be completed, and a design load path identified for each of the RPV missile protection slabs based on ALARP considerations.

316 In terms of lifting of the RPV closure head, EDF and AREVA have identified two loads paths for ALARP consideration. They have described two lift options: Load Path 'A' being a direct vertical lift of the head to the final height, followed by a horizontal translation to the lay down position; and alternatively Load Path 'B' being a short vertical lift, then a short translation to move the head away from the RPV, followed by the final vertical lift to the required height, followed by the necessary translation to the lay down position (including movement over the top of the open RPV below). EDF and AREVA stated that based on the criterion of RPV head 'flyover', the second option was the preferred choice. Through discussion I also commented that in terms of changes in crane movement, this option also was better since the change to the lateral movement from the vertical movement occurred with the head at a much lower height, and a change in demand on the crane was a possible failure point in time. I agree with EDF and AREVA that based on the evidence provided that this Load Path 'B' option is the preferred choice, and as an Assessment Finding (**AF-UKEPR-ME-16**) this should be reflected in appropriate design and safety documentation.

317 EDF and AREVA presented an analysis of the load path / route for the single Fuel Pool Stop Gate between the two gate positions, (i.e. the cask loading pit, and the RPV transfer

pit). EDF and AREVA explained that two load routes were possible, both effectively outside the spent fuel pool area, but for one path the space available between the poolside and the wall was circa 1.9m (Load Path 'A'), whereas for the other path the space was circa 4.4m (Load Path 'B'). Hence the 4.4m space route is considered to be the preferred choice, and as an Assessment Finding (**AF-UKEPR-ME-17**) this should be reflected in appropriate design and safety documentation.

318 I also consider it to be an Assessment Finding (**AF-UKEPR-ME-18)** for a future licensee to ensure that all lifts of nuclear safety significance are identified, and safe load paths are specified through appropriate design and safety documentation, and procedures.

319 I have also questioned EDF and AREVA regarding design for provision of loose article control, although I consider this to predominately be a matter for operational and specifically EMIT consideration. EDF and AREVA have confirmed that this requirement is recognised through their design specifications, and I am satisfied from a GDA perspective.

320 Subject to the assessment findings as described below, I am satisfied with the justification provided in respect of nuclear lifting and design principles for the UK EPR from a GDA perspective against SAP EDR.1.

4.10.2 Findings

AF-UKEPR-ME-14: *The licensee shall ensure the design of all rigging equipment associated with lifts of nuclear safety significance is completed, and in doing so shall systematically review these rigging arrangements to identify faults, and review and implement reasonably practicable improvements to either eliminate such faults by design, or limit their frequency by the provision of engineered protection systems. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-15: *The licensee shall specify the design load paths for the RPV missile protection slabs based on ALARP principles, based on completion of any necessary dropped load consequence studies. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-16: *The licensee shall specify the choice of RPV head lift load path / route based on the ALARP considerations described in the response to RO-UKEPR-052, UK EPR GDA – Management of Nuclear Safety Significant Lifting, ECEMA101802 Revision B). Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-17: *The licensee shall specify the choice of Fuel Pool Stop Gate load path / route based on the ALARP considerations described in the response to RO-UKEPR-052, UK EPR GDA – Management of Nuclear Safety Significant Lifting, ECEMA101802 Revision B). Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the* *appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-18: *The licensee shall ensure that all lifts of nuclear safety significance are identified, and safe load paths are specified through appropriate design and safety documentation, and procedures. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

4.11 Nuclear Ventilation

321 Nuclear ventilation systems play an important role in a NPP in controlling the spread of radioactive contamination in normal and accident conditions, and directing any discharges to suitably filtered routes.

322 During my Step 4 assessment I have selected the following design aspects and systems for assessment:

- Design and Testing of High Efficiency Particulate Arrestor (HEPA) Filtration.
- Emergency Habitability Systems.
- Comparison to Relevant Good Practice.
- Proposed Ventilation Design Changes.

323 I consider the following Safety Assessment Principles to be relevant to this aspect:

- Safety Assessment Principle ECV.2 (Ref. 4) states ‘Nuclear containment and associated systems should be designed to minimise radioactive releases to the environment in normal operation, fault and accident conditions’.
- Safety Assessment Principle ECV.3 (Ref. 4) states ‘The primary means of confining radioactive substance should be by the provision of passive sealed containment systems and intrinsic safety features, in preference to the use of active dynamic systems and components’.
- Safety Assessment Principle AM.1 (Ref. 4) states ‘A nuclear facility should be so designed and operated to ensure that it meets the needs of accident management and emergency preparedness.’

4.11.1 Assessment

4.11.1.1 Design and Testing of HEPA Filtration

324 I decided to assess the detailed design and testing of HEPA filters used within the UK EPR design, since these provide the important safety function of containment of radioactive particulate, specifically in the event of postulated accidents within the NPP.

325 EDF and AREVA provided information regarding testing of HEPA filters, describing factory type tests, followed by batch sample tests as required, TQ-EPR-1438 (Ref. 10).

326 EDF and AREVA provided a significant amount of detail regarding the make up of the aerosol test spray used to determine the filter efficiency, and the associated testing

arrangements and standards. I am satisfied that they demonstrated an adequate depth of understanding in this technical area.

327 In response to questioning EDF and AREVA stated that filters are nominally changed after 4~6 years within French power plants, although the doctrine for the UK EPR has yet to be established. They also stated that no specific problems have been experienced with sealing of their rectangular filter designs. The casings of all HEPA filters and iodine traps for the UK EPR are designed for a lifetime of 60 years.

328 EDF and AREVA stated that dampers are used to adjust and balance the ventilation system performance during system commissioning. I consider that the filter change philosophy should obviate the need for system re-balancing as the filters start to 'clog' during their operational usage. I consider it to be an Assessment Finding that a future licensee establishes an appropriate filter change doctrine for all safety important filters within the nuclear ventilation systems (**AF-UKEPR-ME-19**).

329 In summary, I consider that EDF and AREVA described a mature technology in line with my regulatory expectations against SAP ECV.3, and I do not have concerns in this area.

4.11.1.2 Emergency Habitability Systems

330 I have assessed the provision of HVAC in the Main Control Room in the event of a release of radioactive contamination from the NPP, to ensure that adequate protection is provided to allow the plant operators to undertake any necessary actions.

331 EDF and AREVA provided information on the Main Control Room HVAC design, with specific reference to the provision of iodine filtration.

332 The iodine filtration is not specifically intended to protect from design basis iodine discharges from its own reactor, since this has effectively been eliminated by design due to the use of iodine traps; but is primarily intended to protect against iodine discharged from adjacent plants.

333 The system is capable of operating in ~100% re-cycle mode in the event of external iodine concentrations, with only a slight in-bleed through one of the 50% filters to create the necessary positive pressure. Hence the UK EPR only requires 2 x 50% iodine filtration within the design.

334 Each of the two 50% iodine filters can be aligned with either of two air supply units, to provide the necessary redundancy in the supply system; (note there are four air supply units in total).

335 EDF and AREVA stated that the original consideration had been to provide 4 x 50% capacity iodine filtration, based on the requirement to accommodate maintenance considerations and the Single Failure Criterion. However, subsequent studies had downgraded the iodine filtration system from F1A to F2, based on dose assessment work, such that iodine filtration is only required for severe accident and external hazard circumstances. This philosophy only requires 2 x 50% iodine filtration provision.

336 I questioned EDF and AREVA regarding the reasonable practicability of providing additional filtration, and EDF and AREVA stated that this would be difficult in principle due to space constraints. In response to my questioning, they stated that in the event that only one filter train was available, then the Main Control Room would pressurise, although to a lesser extent than for the full system in operation. However, the extent of

this shortfall could only be determined by on site testing, since the in flow was only required to overcome adventitious leakage, TQ-EPR-1439 (Ref. 10).

337 EDF and AREVA stated that because of the low frequency associated with demands on the iodine filtration system, no specific restrictions are placed on the associated maintenance activity in French practice.

338 The Main Control Room is physically separate from other rooms within the NPP, to minimise the potential for contamination, which EDF and AREVA stated is an improvement on previous French practice.

339 EDF and AREVA stated that previous French power plants only had 1 x 100% iodine filtration provision, and so they considered the provision of 2 x 50% filtration as an improvement over their ‘normal’ practice.

340 The Emergency Control Centre within the NPP is ventilated by the same system as for the Main Control Room, which provides room pressurisation, plus intake HEPA and iodine filtration as demanded, which is in line with my expectations, TQ-EPR-668, (Ref. 10).

341 In summary, and through liaison with my PSA assessment colleagues, I am now satisfied with the explanation provided by EDF and AREVA and the design principles for the Emergency Habitability Systems against SAP AM.1. I consider that adequate protection is afforded by the nuclear ventilation systems to protect the plant operators under the postulated emergency scenarios, taking account of the very low frequencies of these events.

4.11.1.3 Comparison to Relevant Good Practice

342 I have compared the UK EPR ventilation design against UK Relevant Good Practice, including a review of the nuclear ventilation stack design height, and also application of the system design to the UK climatic environment.

343 EDF and AREVA provided information covering comparison to UK Relevant Good Practice in respect of nuclear ventilation, considered to be represented by document ‘An Aid to the Design of Ventilation of Radioactive Areas’, (Ref. 23). I consider that they have undertaken a thorough comparison against this guidance document, TQ-EPR-1443 (Ref. 10).

344 In particular the UK EPR nuclear ventilation design ensures that all potentially radioactive airborne particulate discharges from the NPP are subject to HEPA filtration, both in normal and under fault conditions, which I consider to be a necessary and appropriate design provision. Chemical filters to trap radioactive iodine gas are also used under accident conditions.

345 In respect of dynamic containment within the ventilation systems, EDF and AREVA confirmed that they were in line with UK velocities of between 0.5 and 1.0 m/s for air flows at interfaces between areas of different potential contamination.

346 The UK EPR ventilation systems are designed for the following external air temperatures:

- Maximum 12 hour average temperature as 36 degrees C.
- Highest instantaneous temperature as 42 degrees C.
- Low temperatures as -15 degrees C permanently.

- Extreme low temperature as -25 degrees C for seven days, to -35 degrees C for 6 hours.

347 I also note that the UK EPR is designed for a maritime site, and the above stated temperatures should be reviewed in this context.

348 EDF and AREVA have stated that in the event of temperatures being experienced outside these ranges, then no cliff edge effects are anticipated, however some loss of system performance may be expected. I agree that this is a reasonable assertion.

349 They have also stated that the UK EPR design temperatures for the ventilation system should be reviewed on a site specific basis, once the sites for the proposed UK EPR NPPs have been determined. I agree that this is appropriate, although I also consider this to be an Assessment Finding (**AF-UKEPR-ME-20)**.

350 They have also stated that the nuclear ventilation systems have been designed to account for the UK maritime climate, and in particular stainless steel will be used for the air intake and exhaust grids, the first heaters on the air supply trains, and all the ducts before these first heaters for these air supply trains. I have not identified any concerns in this area.

351 I have questioned EDF and AREVA in respect of the temperatures in the spent fuel pool area, to gain an understanding of the capacity of the nuclear ventilation systems to provide a reasonable working environment for operators, TQ-EPR-550 (Ref. 10). They have stated that the specific ventilation system dedicated to this area has been designed to maintain the pond area temperature at between 20 degrees C and 33 degrees C (in the worst case fuel load). Recognising that this is not a routinely occupied area, and this upper temperature is based on worst case conditions (including outside air temperature), I consider this approach to be reasonable.

352 I also questioned EDF and AREVA in respect of the nuclear ventilation discharge stack height for the UK EPR, to ensure adequate dispersal of radioactive material under normal conditions, but primarily in the event of an accidental release, TQ-EPR-687 (Ref. 10).

353 EDF and AREVA have explained that the detailed design of the stack is a site specific consideration, accounting for the local topography and adjacent buildings, but as a minimum the stack height will need to be equal to the height of the main reactor building (circa 60m). EDF and AREVA have also stated that the stack height for Flamanville 3, as an example, is 64m high. This response is in line with my expectations, and I therefore have no concerns in respect of stack height from a GDA perspective.

354 In summary, EDF and AREVA stated that they generally have equivalent or more stringent criteria for their ventilation systems, and that the OEF of the French fleet underpins their practices. I am satisfied with the information provided by EDF and AREVA against SAP ECV.2, and have not identified any concerns associated with this area.

4.11.1.4 Proposed Ventilation Design Changes

355 I have also assessed the proposed design change to the UK EPR ventilation system, which I understand has resulted from a detailed review of the FA3 ventilation systems undertaken during detailed design, following questions from the French regulatory authorities. I became aware of these changes during the Step 4 process.

356 EDF and AREVA provided a description of the proposed ventilation design changes (CMF-020, Ref. 49) which have resulted from detailed review of the FA3 design. The identified problem is that in the event of certain postulated fault scenarios, including severe accident conditions, leakage from containment penetrations could contaminate peripheral buildings, creating potential direct leaks to the environment, greater radiological contamination for equipment within these buildings, and subsequent accessibility difficulties for plant operators following a severe accident for subsequent remedial operations.

357 EDF and AREVA explained that although all discharges are presently HEPA (and iodine as necessary) filtered via the nuclear ventilation system, this normal discharge route is not qualified for certain accident scenarios, which could result in loss of this dynamic containment. In order to alleviate this, the design change provides connections from the normal ventilation discharge route from containment (Containment Sweep Ventilation System, (EBA) high capacity route), to the higher safety qualified EBA low capacity route, in the event that the containment isolation valves associated with this first system leak or fail. Furthermore, the Fuel Building ventilation system will switch automatically to the EBA low capacity discharge route on containment isolation signal (and / or other specified safety signals), to provide safety qualified HEPA and iodine filtration.

358 EDF and AREVA also explained that additional pre-filters are now added to the annulus ventilation system upstream of the HEPA / iodine filter train, and also the DWL (Controlled Safeguard Building Ventilation System) / EBA ventilation systems upstream of the HEPA / iodine filter trains, to capture potential contamination and hence reduce radiological loading on the other filters, to facilitate post severe accident accessibility.

359 I consider these changes to be rational and reasonable, and expect them to be incorporated into the UK EPR design, as an Assessment Finding (**AF-UKEPR-ME-21**).

4.11.2 Findings

AF-UKEPR-ME-19: *The licensee shall establish an appropriate filter change doctrine for all safety important filters within the nuclear ventilation systems. Target Milestone – fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-20: *The licensee shall verify the site specific design air temperatures and humidity values against those used as the basis for the UK EPR design, to ensure that the nuclear ventilation systems can adequately perform their safety functions. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - delivery to site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-21: *The licensee shall ensure that the proposed modification to the nuclear ventilation system, described as CMF-020 (Confinement – Modification of Ventilation Systems) is fully incorporated into the UK EPR design and safety documentation. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

4.12 Gloveboxes / Cabinets

360 I have undertaken a limited review of gloveboxes and cabinets as part of my assessment, which have the safety function of containment of radioactive substances.

4.12.1 Assessment

361 EDF and AREVA have stated that a small number of gloveboxes are located in the Nuclear Auxiliary Building and the Effluent Treatment Building. They have stated that all the gloveboxes are equipped with HEPA filtration in the air supply duct, and the exhaust duct, and there is also iodine filtration in the exhaust duct to filter iodine releases (with an upstream electrical heater). Gloveboxes are also maintained at a negative pressure with respect to the surrounding room, in line with standard practice.

362 EDF and AREVA have also stated that fume cupboards are used within the UK EPR, but I have not pursued any further detailed information / justification from a Mechanical Engineering perspective. However, fume cupboards are not appropriate for containment of radioactive materials, and I consider it to be an Assessment Finding (**AF-UKEPR-ME-22**) that a future licensee restricts their use to appropriate chemical hazards only.

363 There is a limited requirement for this type of equipment within the UK EPR, with a relatively low nuclear safety significance (excepting ventilation filtration aspects) in the context of an NPP, and it represents a mature technology. I have not identified any concerns in this area as part of my assessment.

4.12.2 Findings

AF-UKEPR-ME-22: *The licensee shall ensure that fume cupboards within the UK EPR are not used for the containment of radioactive substances. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

4.13 Heat Transfer (CCWS and ESWS) and Heat Exchangers

364 I have undertaken an assessment of the heat transfer systems within the UK EPR design, with a specific focus on the Component Cooling Water System (CCWS) and the Essential Services Water System (ESWS). These are the two primary systems for removing nuclear decay heat from the reactor under shut down conditions (control rods inserted), and from the spent fuel pool. I have also extended this assessment line of enquiry to cover the provision of ultimate heat sink (the sea).

365 I also decided to assess the general designs of heat exchangers used within the UK EPR, in terms of their design pedigree, Operational Experience Feedback, and maintenance requirements / practicability.

366 I consider the following Safety Assessment Principles to be relevant to this aspect:

- Safety Assessment Principle EHT.1 (Ref. 4) states ‘Heat transport systems should be designed so that heat can be removed or added as required.’
- Safety Assessment Principle EHT.3 (Ref. 4) states ‘A suitable and sufficient heat sink should be provided.’

- Safety Assessment principle EDR.1 (Ref. 4) states 'Due account should be taken of the need for structures, systems and components important to safety to be designed to be inherently safe or to fail in a safe manner and potential failure modes should be identified, using a formal analysis where appropriate.'

4.13.1 Assessment

4.13.1.1 Component Cooling Water System and Essential Services Water System

367 EDF and AREVA provided information covering the Component Cooling Water System (CCWS) and Essential Services Water System (ESWS).

368 The CCWS comprises four safety classified, geographically separated trains, corresponding to the four electrical trains, and corresponding to the four Residual Heat Removal trains. Within the CCWS there are two common auxiliaries loops; Loop 1 connected to Train 1 or Train 2; and Loop 2 connected to Train 3 or Train 4. The cooling demands within these loops can be switched between the two respective heat exchangers. However, the auxiliaries are themselves split, with the spent fuel pool heat exchanger on a separate loop to allow isolation and maintenance of the other auxiliaries equipment without affecting the spent fuel pool cooling capability.

369 Loop switchover between the heat exchangers is achieved by the use of fast acting butterfly valves in order to limit the transient effects. These valves are based on pneumatic actuation technology, to provide the required rapid actuation time. EDF and AREVA have also stated that a switchover is performed at least twice per month in order to balance the operation of the CCWS pumps, and as such no periodic testing is separately envisaged for these valves.

370 I have questioned EDF and AREVA on the design provision to account for the water hammer phenomenon, which can lead to high pressure transients occurring within fluid systems, leading to the potential for mechanical damage, and ultimately fracture of pipework and fittings, TQ-EPR-1052 (Ref. 10).

371 EDF and AREVA have stated that guidance is available for system designers and operators to avoid water hammer, based on the operating experience from the French fleet of NPPs. Specifically for the CCWS, cast iron is not used within the design, and valve opening and closing times have been selected to limit the induced pressure transients. EDF and AREVA have referred to a specific report to study and justify the CCWS design in respect of water hammer, 'Water Hammer on CCWS' (Ref. 24). In this report EDF and AREVA have stated that the maximum predicted pressure due to water hammer effects is below the system design pressure and the set point of the pressure relief valves, fixed at 14 bar absolute.

372 I consider that EDF and AREVA have described a rational approach, and I am satisfied that they have adequately addressed this phenomenon.

373 I have questioned EDF and AREVA on the provision of make up water to the CCWS system, specifically in light of the fact that there is a single branch connection to the Distribution of Demineralised Reactor Water System (SED), TQ-EPR-1155 (Ref. 10). EDF and AREVA have stated in response that the CCWS is a closed system, and the SED is only envisaged to be used for initial make up, or in the event of small leakage from the CCWS itself. They have also stated that each train of the CCWS is provided with a 27 cubic metre volume expansion tank to absorb the thermal expansion of the CCWS fluid, and to compensate for a CCWS leak before isolation is effected. This tank

is equipped with level measurement, and a drop in level leads to automatic actions to isolate non safety important auxiliaries initially, and then to isolate the less safety important auxiliaries if further level drop occurs. Once the source of the leak is identified then the CCWS train is switched over to the other train of the common pair.

374 In respect of the single connection to the SED, EDF and AREVA have stated that the design is similar to that used in previous French NPPs, and there are no known events that have led to the requirement to modify the design. In the event of failure of the SED connection, EDF and AREVA stated that the CCWS trains are closed systems, and there would be no initial effect on the capability of the CCWS. However, they also state that the loss of this connection would be studied as part of the response to RO-UKEPR-041, raised by the Fault Studies discipline, to address passive single failures. I have reviewed the response to this Regulatory Observation, 'Passive Single Failure Analysis' (Ref. 25), and have not identified any reference to this matter. I consider it to be an Assessment Finding (**AF-UKEPR-ME-23)** that this analysis should be undertaken.

375 EDF and AREVA have described the choice of materials for the ESWS pumps, based on extensive Operating Experience Feedback, including liaison with organisations involved in nuclear marine propulsion. Furthermore they have stated that these pumps, which convey non radioactive fluids, will be subject to a regular monitoring and maintenance regime to identify any unpredicted corrosion / erosion phenomena, to allow for replacement of parts as necessary. For the ESWS pipes, the material is carbon steel with an internal surface coating of neoprene, due to its high corrosion resistance and its capacity to withstand distortion. EDF and AREVA have stated that such pipes will be manufactured to stringent quality criteria, and the pipework is assembled in flanged sections to facilitate the ease of replacement of pipe parts. The minimum thickness of neoprene applied is 4mm, and EDF and AREVA claim that Operational Experience Feedback on similar pipes at Gravelines shows that this thickness is sufficient; i.e. after 25 years of use the minimum thickness measured on the most constrained areas is 3.6mm.

376 For the CCWS the UK EPR uses carbon steel for pump casings, pipework and other equipment, based on the use of demineralised water subject to a suitable chemical dosing regime. In the limited areas where raw water is used then suitably coated pipework is specified, or specific corrosion resistant materials selected. However EDF and AREVA have not identified the practicability of inspecting and / or replacing detrimentally affected sections of the CCWS in respect of corrosion. I consider it to be an Assessment Finding (**AF-UKEPR-ME-24)** that this should be undertaken and any necessary ALARP improvements which are identified, implemented.

377 Notwithstanding this Assessment Finding, I am satisfied with the explanations provided in respect of this line of enquiry, for design principles and materials of construction for both the ESWS and the CCWS against SAPs EHT.1 and EDR.1.

4.13.1.2 Ultimate Heat Sink

378 EDF and AREVA provided information on the provision of the ultimate heat sink for the UK EPR design, which has been an important line of enquiry to follow the cooling provision within the heat transfer systems to its ultimate destination, i.e. the sea.

379 The UK EPR utilises a forebay structure which takes in-feed sea water from the intake tunnels, and which provides a filtered water supply to both the Essential Service Water

System (ESWS), and the Ultimate Cooling Water System (UCWS). The intake structure connects to two large diameter tunnels, which extract water from the sea via a number of vertical shafts connected to intake heads. The detailed design of the configuration is understood to be a site specific matter, but I consider that the principle lies within the scope of GDA. EDF and AREVA explained that although the height of water within the forebay structure will vary with the tide, the design is such that it is filled with an adequate water volume on a continuous basis.

380 Water from the forebay structure is initially filtered by the pre-filtering system comprising grids and trash rakes, which trap debris within the water and then remove it for disposal to ensure a relatively clear flow of water to the pump house is provided. The water then passes into the pump house, which comprises four separate trains of the Circulating Water Filtration System; two inner trains provided with large drum screens which also provide feedwater for condenser cooling as well as the ESWS / UCWS, and two outer trains with band screens which provide water to the ESWS / UCWS only. EDF and AREVA stated that the pre-filtering system was categorised / classified as Cat C / Class 3 against the UK methodology, which is generally in line with my expectations; and the drum and band screens as being Cat B / Class 2 against the UK methodology, which again is generally in line with my expectations due to the higher importance of these systems. I have confirmed that this is reflected in the latest safety categorisation / classification documentation (Ref. 32).

381 In response to my questions EDF and AREVA stated that drum screens are required for the central trains due to the very high water flows required for condenser cooling, and band screens are provided for the outer trains since they are considered to be more physically robust, and their separate design provides a degree of diversity. I consider this to be a rational approach.

382 In respect of the drum and band screen, the speed of the mechanical systems and cleaning pressure are automatically controlled to optimise evacuation of debris, based on head loss measurement. In the event of considerable head loss across either the drum or band screens, the downstream pumps are tripped for the non essential water supplies (including for the condenser cooling), which allows the exit side of the screens to recover their head, to provide a sufficient flow of filtered water to the ESWS / UCWS. This design takes into account OEF from the French Chooz NPP where there was no automatic trip under these circumstances, (simply an alarm in the control room).

383 The downstream ESWS design comprises a dedicated pump for each train, flow measurement, maintenance isolation valves, a shell fish screen, and a U tube heat exchanger to the Component Cooling Water System. This system is categorised / classified as Cat A / Class 1, which is in line with my expectations. EDF and AREVA stated that in service maintenance is permitted on one train only, in line with four train philosophy of the plant. A common suction header is provided for all four trains, to ensure that each of the four ESWS / CCWS trains can be connected to each of the four Circulating Water Filtration System trains.

384 EDF and AREVA have also stated that proliferation of marine organisms within the ESWS is prevented by the design and operation of the Circulation Water Treatment system.

385 The Ultimate Cooling Water System comprises two trains, each utilising a dedicated pump, a shell fish screen, flow measurement, isolation valves, and a U-tube heat

exchanger. A common header arrangement is also provided so that each train can be connected to each of the four Circulating Water Filtration System trains.

386 The UK EPR design incorporates two diversification pipes, which can be used to provide water from the outfall structure; one pipe able to feed the ESWS and the other the UCWS. EDF and AREVA have confirmed that both these pipes are within the scope of the UK EPR GDA, and they have been referenced in the PCSR (Ref. 13), and in the response to the related Technical Query, TQ-EPR-1006 (Ref. 10). EDF and AREVA stated that these diversification pipes are F2 classified (system architecture), and they are intended to allow continued use of the ESWS and the UCWS in the event that the pumping station becomes unavailable; effectively by making the outfall structure into the source of water, and converting the forebay structure into the new outfall by manually realigning valves. In response to questions EDF and AREVA stated that they were not intended to allow the heat transfer systems to operate in ‘closed loop’ mode, whereby water entering the outfall structure from the plant could be fed back into the ESWS and UCWS, thereby removing the reliance on the sea based ultimate heat sink, (albeit based on a degraded heat sink capability). Following discussion, EDF and AREVA agreed that the system would have some capability in this respect, but this had not been studied or quantified. I consider it important to understand the capability in this respect as a design activity, as a defence in depth provision, since cooling of the NPP is a key high level safety function. I consider it to be an Assessment Finding (**AF-UKEPR-ME-25**) that this capability should be understood and defined.

387 Notwithstanding this Assessment Finding, I consider that EDF and AREVA have described a rational design approach in respect of the ultimate heat sink provision, which has benefitted as appropriate from OEF. I am therefore content from a GDA perspective against SAP EHT.3.

4.13.1.3 Heat Exchanger Designs

388 The Residual Heat Removal System (RHRS) heat exchangers are of conventional shell and tube design, and are of similar design and manufacture to those used in previous NPPs in France and Germany. The Spent Fuel Pool (SFP) heat exchangers are also of shell and tube design, albeit using a two shell design to achieve the correct thermal performance. EDF and AREVA have stated that the RHRS and SFP heat exchangers are of the same overall design to those used in the N4 NPPs, and more generally the French fleet, with no negative operational experience feedback recorded to date. I have not identified any concerns in this area from a GDA perspective.

389 EDF and AREVA explained the design requirements associated with the sizing of the CCWS / ESWS heat exchanger, in terms of cooling rate and flow rate under normal and fault conditions.

390 The Essential Services Water System has the function to provide cooling to the CCWS under normal plant operating conditions, normal cooldown / shutdown conditions, and plant design basis accident conditions. Each of the four ESWS trains comprises 1 pump, instrumentation, and the CCWS / ESWS shell and tube heat exchanger with its integral cleaning device. The ESWS sea water is on the tube side of the heat exchanger, and the CCWS is on the shell side.

391 On the ESWS tube side, a cleaning device operates continuously, based on Konvoi technology. This is based on the use of soft foam type balls which are fed around the

tube circuit via an injection line and recovery device, which clean the tubes by abrasion. EDF and AREVA explained that the heat exchanger used titanium as a material of construction on the tube side due to the sea water corrosive environment, or carbon steel with titanium cladding, or carbon steel with an epoxy coating.

392 EDF and AREVA explained the design criteria in terms of heat transfer, incorporating margin on the heat transfer area and on the pressure drop, with a fouling rate imposed for each type of fluid, and limitations on the selection of materials, tube thicknesses and fluid speeds.

393 Although the heat exchanger maintenance plan has not yet been determined, EDF and AREVA explained that it would account for the following main points:

- Corrosion assessment.
- Seal replacement.
- Cleaning.
- Maintenance of the ball cleaning device.

394 EDF and AREVA stated that there were no general radiological issues associated with maintenance of the CCWS / EWCS heat exchanger, since both sides were not contaminated, and in response to my question they stated that the CCWS does have contamination monitoring to support this. EDF and AREVA confirmed the design life of the heat exchanger as 60 years.

395 They explained that the CCWS / ESWS had benefitted from Operational Experience Feedback (OEF) from the Konvoi designs and French power plants, through automatic control of fluid flow through the heat exchanger, limitation on hot fluid to cold fluid temperature differences, and design of the tube side ball cleaning device.

396 EDF and AREVA stated that as part of their design selection process, they had considered the benefits and dis-benefits of using an alternative plate and frame heat exchanger design. The decision had been taken to adopt the shell and tube design, largely driven by the benefits in respect of fouling tolerance due to the interaction with sea water, noting the beneficial adoption of a tube cleaning device. This design selection has good operating experience on existing French plants, with no adverse OEF from circa 30 years of operating.

397 I consider that EDF and AREVA provided a good demonstration of a rational design approach in respect of the UK EPR heat exchanger designs, which has benefitted as appropriate from OEF. I am satisfied with the system design and design principles described, from a Mechanical Engineering GDA perspective against SAP EDR.1.

4.13.2 Findings

AF-UKEPR-ME-23: *The licensee shall ensure that the analysis of a passive failure of the single branch connection of the Distribution of Demineralised Reactor Water System to the CCWS is undertaken, and any resulting findings are incorporated into all necessary design and safety documentation. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-24: *The licensee shall assess the practicability of inspecting and / or replacing detrimentally affected sections of the CCWS in respect of corrosion, and implement any necessary ALARP improvements which are identified. Target milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-25: *The licensee shall quantify the heat transfer capability of the ultimate heat sink to operate in closed loop mode, specifically by the use of the UCWS diversification pipe, and develop any necessary operating instructions to provide a capability in this scenario. Target milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

4.14 Diesel Generators

398 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. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment Principle ESS.1 states ‘All nuclear facilities should be provided with safety systems that reduce the frequency or limit the consequences of fault sequences, and that achieve and maintain a defined safe state.’

399 My Step 3 assessment determined that EDF and AREVA assigned a safety claim on the diesel generators. During my Step 4 process I have undertaken sample assessments of the available evidence that supports the diesel generator safety functions, EMIT, and how Operational Experience Feedback is considered within the design.

4.14.1 Assessment

400 The UK EPR system design uses 4 main diesels, and 2 ultimate diesels. The diesels are specified to provide 72 hours of operation (for the engines), which is based on fuel provision, and so this timescale can be extended by provision of additional fuel. For the main diesels, 180 cubic metres of fuel is provided for each unit.

401 I have discussed the provision of margin in the sizing of the diesels, and EDF and AREVA have stated that the design allows for a capacity margin of 10% above the requirements, as a residual margin at the end of the design process; (noting that design margins may also be used due to the iterative nature of the design process). I consider this to be a reasonable approach.

402 The diesel generators are housed in dedicated areas in the NPP, which comprise two buildings, each housing two main engines and one ultimate diesel. Each diesel is fuelled separately, and EDF and AREVA have stated that the engines are of the next generation designs, but are the same size as for the N4 series of machines.

403 The diesels are started with the aid of compressed air, which has a capacity to start a diesel up to ten times, prior to requiring to be refilled and there are two vessels dedicated to each diesel unit.

404 My assessment of Classification of SSCs (Ref. 32) provides the evidence that the diesel units are assigned with a safety Class 1 for the main four diesels and a safety Class 3 for the two ultimate diesel generators units. The assignment of a Class 3 against the ultimate diesel units is lower than my initial expectations. I consider it to be an Assessment Finding (**AF-UKEPR-ME-26)** that the safety class of the ultimate diesels is reviewed, since I consider that the declared assignment does not correctly reflect the importance of this system to safety. However, I recognise that this assignment needs to be consistent with the methodology adopted against the UK expectations in this area.

405 My assessment of OEF from the IAEA Incident Reporting System database identifies that the failure to start diesel generators on demand is a common theme. In response to questions, EDF and AREVA stated that they have an expert group for diesel generators which covers the operating experience of these machines across the French fleet of circa 50 NPPs. Each diesel is the subject of routine testing, which entails running each unit for 2 hours every 2 months at 25% load to verify the safety functional requirement, and that they will be operable on demand.

406 TQ-EPR-1446 (Ref. 10) response provides details of the process for the capture of OEF undertaken for the French fleet of NPPs.

407 The process captures Operational Experience Feedback from three discreet aspects:

- Design, procurement and quality management systems - feedback is shared between the various projects via monthly meetings. The monthly meetings are attended by representatives from the design, procurement, engineering, and site operations that represent the whole of the French fleet of NPPs.
- Operational Experience Feedback from the French fleet of nuclear power plants is managed by EDF Operations Division who uses a national database that records operating events. Two meetings per year are undertaken with the prime role being to discuss the events and any recommended changes in carrying out maintenance activities.
- International feedback from WANO and INPO is typically captured by corporate events and IAEA feedback, which is managed by the Operations Division.

408 OEF can be incorporated into the plant design at any stage of an NPP life cycle, for example the design concept phase, manufacturing phase, and commissioning through to the operational phase. Implementation of a change is managed following the design change process.

409 Examples of design changes implemented into the diesel generator system design, as a direct result of Operational Experience Feedback are described as follows:

410 Example 1 - Diesel generator cooling system and connecting pipework exposed to the external environment:

- Operational experience identified the cooling system equipment and interfacing pipework suffered excessive corrosion when located at a coastal site.
- The UK EPR design now incorporates the equipment within the diesel generator building, thus providing increased protection from an external coastal environment.

- Responses to my questions identified the revised location did not affect the ventilation design or requirements, as the equipment is located within a natural venting room.

411 Example 2 - Diesel generator fuel injection pump:

- Operational experience indicated lubrication issues on the mechanical connections to the injection pump.
- The design now requires the fitting of self lubricating bushings, which are also the subject of a routine maintenance inspection.
- Responses to questions identified this requirement is specific to nuclear applications due to the intermittent use of the equipment.

412 Example 3 - Diesel generator regulator valves:

- Operational experience now requires the valves to be of a thermo static type, which increases the availability of the diesel power source. These valves regulate the high temperature water, low temperature water, oil and air.

413 Example 4 - Diesel generator injection pump:

- In order to avoid air settling and becoming trapped, the fuel injection system now incorporates a valve that purges the line of air, which allows fuel to be injected, allowing the startup of the diesel generator with an increased reliability.

414 EDF and AREVA confirmed that all four design improvements are incorporated into the UK EPR as part of the reference baseline design.

415 In response to questions, EDF and AREVA have further clarified the design philosophy and detailed design in respect of the two tank diesel fuelling system.

416 They stated that the size of the main fuel storage tank of $180m^3$ is determined by the building layout constraints. Seismic considerations position the storage tank at a level lower than the main diesel generators. This configuration may lead to aeration of the injection system during the periods when the motor is in its standby mode. For this reason the design incorporates a smaller intermediate tank, which is continuously fed from the main storage tank and is used to feed fuel to the diesel generators. This tank is located at the 8.1m level to allow fuel to be fed by gravity to the diesel units that are located at a lower level. It supports the reliability of the system to start on demand. The slight overpressure avoids aeration of the injection system and provides an acceptable supply for the pump booster.

417 The intermediate tank sizing has a capacity of $4m^3$ to supply fuel to the diesel generator unit for 2 hrs continuous use at full power. The existing French fleet N4 plant design parameter is for 1 hr continuous use at full power. This increased time parameter provides operators an opportunity to carry out other tasks if deemed necessary.

418 Fuel transfer from the main storage tank is ensured via a piping arrangement that incorporates a redundant electric pump and instrumentation to manage the filling process and fuel tank levels.

419 The proposed maintenance regime associated with the diesel generators, only allows one machine being taken out of service at any one time, with a total annual cumulative downtime limited to 28 days for all diesels, and a maximum limit of 14 days for any one machine. However, following my assessment of the diesels, I consider there is limited evidence showing how the diesel EMIT requirements are transferred into a Plant

Maintenance Schedule. I consider this to be an Assessment Finding **(AF-UKEPR-ME-27**) (although the topic is also covered generically under cross-cutting Regulatory Observation RO-UKEPR-055).

420 The seismic qualification requirement for each diesel design is based on a combination of analysis and physical testing. I have not pursued this area in detail, but consider the approach to be rational from a Mechanical Engineering perspective.

421 Considering the safety importance of the diesel generator units, I was interested to understand how EDF and AREVA are taking into account the amendment to the Motor Fuel (Composition and Content) Regulations 1999, which is to be implemented under EU Directive 2009/30/EC. The amendment is concerned with implementing more stringent control of fuel parameters, which have an environmental impact. I consider it to be of particular interest to understand the impact of the increased use of biofuels and the consequential effects on the performance of the diesel generators.

422 Response TQ-EPR-1182 (Ref.10) states the detailed procurement specification will be defined outside the GDA and will be part of Phase 2 and site licensing. In addition the unit motors and auxiliaries are designed to comply with the characteristics as defined by the supplier.

423 EDF and AREVA confirmed that the specifications for the FA3 plant do not specifically consider mixed fuel as a specific design requirement. However, the design can allow the use of mixed fuel without any degradation of the engine performance within a limited range, for example the design limitation for biofuel is 10%. I consider the use of fuel outside this design constraint may impact the ability to start a unit on demand and introduce engine malfunction.

424 They advised that the fuel requires an appropriate viscosity at the injection pump and the introduction of biofuel, mixed or contaminated fuel may lead to the fuel viscosity moving outside of specification, so impairing a diesel unit performance. However EDF and AREVA claim the fuel viscosity can be maintained within specification by the addition of a water heat exchanger.

425 In summary, my assessment has identified that the UK EPR diesel systems currently do not adequately take into account the regulation amendment in respect of fuels, (Motor Fuel (Composition and Content) Regulations 1999) and I consider this to be an Assessment Finding (**AF-UKEPR-ME-28**); a future licensee is required to provide further evidence that adequate consideration is given to the applicable legislation.

426 Notwithstanding this Assessment Finding EDF and AREVA have provided good evidence in respect of their diesel generator designs, and in particular the review of appropriate OEF and associated design improvements.

4.14.2 Findings

AF-UKEPR-ME-26: *The licensee shall ensure that the safety class of the ultimate diesels is reviewed and justified, since I consider that the declared Class 3 assignment does not correctly reflect the importance of this equipment to safety. However, I recognise that this assignment needs to be consistent with the methodology adopted against the UK expectations in this area. Target Milestone – install diesel generators complete as this is the appropriate point when sufficient* *evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-27: *The licensee shall ensure that the diesel EMIT requirements are adequately transferred into the Plant Maintenance Schedule, (although the topic is also covered generically under cross-cutting Regulatory Observation RO-UKEPR-055). Target Milestone – fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-28: *The licensee shall ensure that the UK EPR diesel systems adequately take into account the regulation amendment in respect of fuels, (Motor Fuel (Composition and Content) Regulations 1999), in terms of meeting their safety functional requirements. Target Milestone – install diesel generators complete as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

4.15 Spent Fuel Handling

427 I reviewed the anti-siphoning features of the spent fuel pool, since this is an important consideration to ensure that the spent fuel pool maintains an adequate coverage of treated water to provide cooling and shielding for the spent fuel.

428 I have also looked at the recent design change covering provision of pool cooling water within the UK EPR.

429 I also decided to review the spent fuel transfer facility associated with the UK EPR design, since this utilises a bottom loading philosophy, via a cask loading pit adjacent to the spent fuel pool, in line with the design for the latest design for the French N4 plants. I visited the Chooz NPP in North Eastern France as part of this assessment activity.

430 I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment principle EDR.1 (Ref. 4) states ‘Due account should be taken of the need for structures, systems and components important to safety to be designed to be inherently safe or to fail in a safe manner and potential failure modes should be identified, using a formal analysis where appropriate.’

4.15.1 Assessment

4.15.1.1 Spent Fuel Pool Anti-siphoning Design

431 EDF and AREVA provided information describing the anti-siphoning system applied to the spent fuel pool, and which is connected to the cooling system suction and discharge lines, and the purification lines; except for the third cooling train. EDF and AREVA explained that 80mm nominal diameter pipes are used for the cooling lines, and 50mm nominal diameter pipes for the purification lines.

432 During discussion EDF and AREVA explained that no anti-siphoning system was provided on the third cooling train since the balance of argument was to maintain this system’s functionality to provide cooling. This was compensated for in the design, since this line is normally isolated with motorised valves, and the length of pipe which cannot be isolated is designed to enhanced standards.

433 The anti-siphoning system is designed to passively terminate an accidental draining of the spent fuel pool following breakage of cooling and / or purification lines. The anti-siphoning system is designed to terminate the flow at a water level 50mm below the bottom of the pipe where it penetrates into the spent fuel pool. In response to questions, EDF and AREVA stated that the anti-siphoning system was sized and designed using established formulae, but levels will be tested and validated during commissioning.

434 In summary, I am satisfied with the evidence and explanations provided by EDF and AREVA for this simple and established technology against SAP EDR.1.

4.15.1.2 Pool Cooling Water Design Change

435 EDF and AREVA explained the design change CMF-004 (Ref. 50) which has been identified for this aspect (which is integrated into the UK EPR design), which had been initiated by the French safety authorities, TQ-EPR-1448 (Ref. 10). This is summarised as follows:

- Provision of hand operated valves on the cooling train suction pipes for accidental draining interruption for the spent fuel pool. EDF and AREVA explained that for the two main cooling trains, two manual valves are provided; whereas for the third cooling train, one motorised and one manual valve are provided. It should also be noted that check valves are provided in the discharge lines of the cooling trains, to protect against siphoning effects in this leg.
- Provision of a drainage line from the instrument lance compartment. This compartment (pool) is adjacent to the reactor building pool (inside containment), and this line allows water to be supplied to the IRWST in the event of an emergency requirement during re-fuelling operations.
- Provision of a reactor cavity overflow line, which in an emergency can be used to fill the IRWST from the reactor building pool, and thus allow the IRWST water to be used in the fuel cooling circuit.
- Provision of automatic closure of suction motorised valves, (valves already in place), in case of low levels detected in the reactor building pools.

436 I have not identified any issues of concern regarding the Mechanical Engineering features of this design against SAP EDR.1. I consider that the technology for the design and operation of such facilities is well established, and EDF and AREVA have taken account of Operational Experience Feedback to improve their design proposal.

4.15.1.3 Spent Fuel Transfer Facility

4.15.1.3.1 Observations from Operating NPP

437 I visited the Chooz Nuclear Power Plant in north-eastern France, which comprises two N4 PWR stations, in order to view the spent fuel handling design, and specifically the export facility design, which is essentially the same as that proposed for the UK EPR.

438 During the visit to Chooz Unit 1 I observed two new Fuel Assemblies (FA) being received and loaded into the spent fuel pool. The two FAs were contained in a single transport container, which contained integral dampers and vibration monitoring equipment to determine if the FAs had been subjected to shock loading during transit. The FA transport container generally comprised an outer and an inner container; the outer lid was

removed and placed on temporary trestles in the receipt area using the auxiliary crane. I noted that space was limited in this area, although adequate for the operation. The inner container was then lifted to a vertical position using the auxiliary crane; using a ‘foot / ram’ within the outer container shell, and a pivot, both of which were integral to the lifting arrangement to maintain the inner container in a safe and stable vertical position. Once the inner container was lifted vertically, a FA was removed and then supported by the auxiliary crane. It was then visually examined, using a hand held torch for any debris between the fuel pins, or for signs of damage. The FA was rotated through 90 degrees to facilitate this, and examined along its full length. I noted the use of temporary scaffolding associated with this operation, to allow the operator to examine the FA from a close distance. EDF and AREVA subsequently confirmed that no such temporary scaffolding is required for new fuel examination for the UK EPR design. They also confirmed that the use of reprocessed uranium for new fuel is considered to be outside the scope of GDA, and therefore no specific radiation protection features are presented within the UK EPR design, to mitigate this hazard, TQ-EPR-755 (Ref. 10).

439 I then observed the FA being moved by the auxiliary crane to the spent fuel pool area, and then lowered into the pool through the funnel of the new fuel elevator, which then lowered the FA down into the pool. I understand that an important criteria here is not to wet the hook of the auxiliary crane to prevent it becoming contaminated, and this was achieved without difficulty.

440 The FA was then successfully moved to the allocated space in the FA storage rack without difficulty, using the Fuel Handling Machine, which comprised a standard EOT crane type with a long square section tool arrangement. I noted that the equivalent crane for the UK EPR will be different, operating from a lower level with a mast type arrangement.

441 EDF and AREVA explained that the FA positioning was controlled by the crane operator, although there were interlocks to prevent FA misplacement. I also observed pool fuel rack position markings on the crane and the building walls, which provided an additional visual check, to indicate the location of the fuel being handled by the Fuel Handling Machine.

442 Following the initial visit to the Unit 1 fuel pond area, the station staff provided a short explanation covering the transfer of the spent fuel assemblies out of the station. This described the spent fuel cask, and its associated cooling requirements; (noting that cooling is enhanced in the horizontal position due to greater natural convection effects). EDF and AREVA explained that external forced cooling is required if the cask temperature exceeds 80°C, (which is achieved via an annulus arrangement integral to the cask).

443 EDF and AREVA described the requirement to precisely level the spent fuel cask to correctly interface with the underside of the cask loading pit, and provide a leak tight seal. The spent fuel cask is transferred into the loading bay area on a rail mounted bogie arrangement, and is held in a substantial fixture. The cask loading pit interface area comprises two position stations; one for removing the final cask biological lid, and the other for connecting to the underside of the pit. The cask is then filled with water, and the injected water also fills large standpipes which connect to the cask and project up into the cask loading pit. EDF and AREVA explained that the filling continues until these standpipes are filled to the level of the pool water above. This then ensures that the cask is full of water, and also that the pressure within the cask is equalised with the bottom of the pit, allowing the hatch-door to be opened.

444 In response to questions, EDF and AREVA explained that the hatch cover in its closed position is locked in place by a mechanical locking feature which slides on top of the door, and wedges into two catch arrangements, one on either side. EDF and AREVA explained that the hatch cover is opened and closed using special tools, and locally controlled from the top of the associated pit.

445 I questioned EDF and AREVA that when the cask is connected to the cask loading pit, then the cask becomes part of the spent fuel pool containment structure, and associated failure could lead to emptying of the pool. This point was understood, but they explained that the penetrations in the cask are small, thus limiting any potential leakage; furthermore connecting pipework has double isolations. They also stated that any leakage would be detected by pool level and pressure detection.

446 I visited Unit 2 of the Chooz station, and viewed a spent fuel cask, and transfer trolley in the loading bay area, and was also able to view the loading pit from the underside. EDF and AREVA explained that there is a double walled seal, incorporating a bellows, to effect the interface between the top of the cask and the underside of the cask loading pit. They also stated that this seal is externally pressurised to prevent leakage. EDF and AREVA explained that the cask transfer trolley / fixture is secured in position to protect against seismic events, and I was able to observe the fixing points in the adjacent walls.

447 EDF and AREVA stated that the spent fuel cask transfer trolley proposed for the UK EPR would be essentially the same as that used at Chooz, the only significant proposed change being more biological shielding for operators at the top of the trolley.

448 I then viewed the cask loading pit from above, and noted the penstock and pool sluice gate which provide a double barrier between the spent fuel pool and the pit. However, EDF and AREVA stated that both doors are open during spent fuel cask loading operations. In response to questions, EDF and AREVA stated that the door, (the penstock having been removed), could be closed by motorised action in circa 2 minutes, and in the event of motor failure, the motor could be manually removed and the door readily closed by manual action, (although this would clearly take longer, but once the operator was at the location, a time of circa 5 minutes would be achieved). I also observed that there is a sill at the pit to spent fuel pool interface, such that the spent fuel pool could not drain below this level in the event of a complete pit emptying, and hence could not uncover the spent fuel.

449 EDF and AREVA explained that for the UK EPR both the cask loading pit and the RPV transfer pit would have a penstock and sluice gate, (whereas in Chooz, the RPV transfer pit only has a sluice gate, without a penstock). EDF and AREVA stated that according to their present rules, two barriers are now generally required if there is water and air on adjacent sides.

450 The design includes an alarm that warns the operator if the cask is not adequately aligned with the bottom of the cask loading pit. On alarm, the operator realigns the cask to the bottom of the penetration to ensure an adequate seal is achieved.

451 EDF and AREVA stated that once the spent fuel cask was filled with FAs, the water is emptied, and the cask filled with helium for transport.

452 I recognise the benefit of this under pool spent fuel transfer technique, which avoids the need for heavy cask lifting over the pool, and also the need for cask decontamination. Although the physical arrangements require a degree of Mechanical Engineering

complexity, I did not identify from this visit any issues which would prevent this being acceptable as part of the UK EPR design.

4.15.1.3.2 Engineering Assessment

453 EDF and AREVA provided further information covering the Spent Fuel Cask Transfer Facility (SFCTF) following my visit to the Chooz N4 nuclear power station. This information covered;

- The purpose and role of the SFCTF.
- The safety functional requirements and lifting equipment description.
- Maintenance and inspection.

454 The role of the SFCTF covers the delivery, preparation and opening of the cask; the cask is then docked to the loading pit via the penetration and loaded with spent Fuel Assemblies (FA); the cask is then closed, conditioned and prepared prior to transport out of the building. The SFCTF comprises the 'DMK' trolley which is rail mounted, and which incorporates a fluid circuit to ensure adequate cooling of the cask. Automatic tasks are controlled and monitored from the control room; which is situated adjacent to the SFCTF.

455 In the handling operation area the fluid, electrical and I&C connections are made to the trolley, the cask's cover is removed, the biological lid is loosened and the cask is filled with water. The cask is then moved to the biological lid handling station where this lid is removed. The cask is then moved under the penetration, and connected via a sealing device, which includes a double walled bellows interface section.

456 In response to my questions, EDF and AREVA explained that incorrect movements are prevented by electrical interlocks; motor driven valves close in the event of loss of power, and mechanical components and valves incorporate emergency controls. Furthermore leaks from the cask and / or the loading pit are detected by sensors with threshold alarms.

457 The safety requirements associated with the SFCTF are summarised as follows:

- Cooling the fuel in the cask before conditioning.
- Ensuring no de-watering of the fuel assemblies before complete closure of the cask.
- Prevention of the cask being dropped during handling, even in the event of a Design Basis Earthquake.

458 The back-up cooling facilities for the cask, (in addition to the water in the cask), comprise cooling by the cooling skirt (which is an annulus partially surrounding the cask, and is integral to the trolley), using demineralised water, then water from the fire protection system, and finally emergency cooling by gravity draining of the cask loading pit. EDF and AREVA clarified that this annulus cooling would only be required if the cask was not connected to the loading pit directly, since in that case natural circulation ensures adequate cooling.

459 EDF and AREVA gave a detailed engineering description of the penetration to the cask loading pit, and described the double barrier concept as applied to this feature. Specifically this incorporates a double walled bellows, and double seals, including specific pipes to monitor the inter-space between the barriers, plus the double seal features on the top and bottom flange details, including monitoring pipes. The pipes

connected to the penetration have small diameters, with a maximum nominal diameter of 40mm to limit a worst credible leak based on pipework failure. EDF and AREVA explained that in the event of leakage, it will be possible to close adjacent valves, the penetration covers, and ultimately the spent fuel pool sluice gate.

460 In response to my questions, EDF and AREVA stated that the stainless steel double bellows was designed for a 60 year plant lifetime, although replacement of parts, or the whole assembly, was practicable under heavy maintenance; noting that the pit above can be empty with the double barriers of penstock and sluice gate in place.

461 EDF and AREVA stated that a key benefit of their bottom loading design is the avoidance of the hazard of cask drop, (noting that the cask weighs approximately 130 tonnes).

462 Operations of the SFCTF are controlled by an operational Programmable Logic Controller (PLC), and also monitored by an independent PLC, which monitors all internal and external interlocks, and which can block orders from the operational PLC.

463 In response to my question, EDF and AREVA stated that access to the SFCTF is prevented during fuel loading operations by administrative control, noting that the control room is effectively adjacent to the facility. However, anticipated dose levels would only be moderate and access would be practicable if necessary, albeit with time constraints.

464 EDF and AREVA stated that this SFCTF would be provided fully designed, commissioned and installed for the UK EPR. They explained that in France the cask design is changing, but they had been in liaison with this process, and the trolley and pit penetration have taken these changes into account, i.e. the UK EPR design works for the existing design of French cask, and will work for the new design of French cask. I consider this to be reasonable and acceptable from a Mechanical Engineering GDA perspective for the UK, through liaison with my waste and decommissioning assessment colleagues. Nevertheless, I consider it to be an Assessment Finding (**AF-UKEPR-ME-29**) that a future licensee ensures that the Spent Fuel Cask Transfer Facility mechanical interface is adequate to ensure transfer of the spent fuel out from the NPP.

465 EDF and AREVA provided further information relating to the integrity of the cask loading pit during operation. The pit is usually empty during normal operation (and refuelling operations). A sluice gate and penstock (Fuel Pool Stop gate – i.e. moveable dam structure) separate the cask loading pit from the spent fuel pool.

466 In response to my questions, EDF and AREVA provided the following information:

- At present, loss of spent fuel inventory is not considered in either the DBA or the PSA, justified on the basis that the associated engineering features are designed and operated for all foreseeable operating conditions; spent fuel assembly transfers are time limited operations; and when no transfers are taking place then the frequency of draining due to leakage is considered very low or negligible.
- The penstock is classified as seismic classification SC1, and the sluice gate as seismic classification SC2. In response to my question, EDF and AREVA stated that the corollary of this was that if the sluice gate was open during a seismic event, you would not be able to close it; however they stated that this has been accounted for in the safety analysis.
- The cask loading pit penetration has the following classifications based on the EDF and AREVA methodology: functional classification F2, mechanical (containment) classification M2, seismic classification SC1.

- Leak tightness is tested following closure of the pit hatch cover; and hatch cover opening and closing is only undertaken with the sluice gate closed.
- Loss of water due to any leakage could be made up by use of the classified Fire Fighting Water Supply System, or by the Nuclear Island Demineralised Water Distribution System.

467 In response to my query regarding the worst credible hazard of spent fuel pool emptying, EDF and AREVA stated that in the event of a worst case 40mm pipe break, and the cask loading pit connected to the spent fuel pool, the drop in pool level would be 43 cm per hour, corresponding to a volume loss of 60 cubic metres per hour; which compares to the capacity of the fire water system of 150 cubic metres per hour. EDF and AREVA explained that this was a worst case bounding situation.

468 I have questioned EDF and AREVA regarding the actions to be undertaken in the event of faults occurring during spent fuel transfer using the SFCTF, TQ-EPR-911 (Ref. 10). I have specifically questioned what actions would be undertaken in the following fault scenarios:

- Leak from cask detected.
- Leak from penetration detected.
- Seismic event.
- Mechanical / control failure of the spent fuel mast.
- Loss of electrical supply to the spent fuel mast.
- Loss of divisional electrical supply.

I have reviewed the response, and have been satisfied from a Mechanical Engineering perspective that sufficient equipment, and reasonable timescales, are available to recover the situation to a safe state.

469 EDF and AREVA stated that there is only one penstock for the UK EPR design, which is shared between the cask loading pit, and the pit which is used for refuelling / defueling the RPV. I have discussed with EDF and AREVA the use of engineering sequence diagrams as part of the design process where mechanical handling and transfers are required. They have subsequently provided an engineering sequence diagram to describe the operations within the spent fuel pool area TQ-EPR-719 (Ref. 10), in terms of door and penstock positions, and water levels, which was of assistance to my assessment.

470 I have not identified any issues of significant concern regarding the spent fuel transfer route from a Mechanical Engineering perspective, although I am aware that further work by EDF and AREVA is required in respect of the fault studies justification, and this may result in further Mechanical Engineering confirmatory assessment as a derivative exercise, which I have captured as an Assessment Finding (**AF-UKEPR-ME-30**). Notwithstanding this, I am satisfied with the design principles from a Mechanical Engineering perspective for GDA against SAP EDR.1.

4.15.2 Findings

AF-UKEPR-ME-29: *The licensee shall ensure that the Spent Fuel Cask Transfer Facility mechanical interface is adequate to ensure transfer of the spent fuel out* *from the NPP. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-30: *The licensee shall ensure that the output from the Regulatory Observation from the Fault Studies discipline (RO-UKEPR-075), relating to the safety case justification covering the spent fuel pool, is reviewed to ensure that all mechanical items important to safety are covered by adequate safety function categorisation and classification, and systems and equipment are specified accordingly. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

4.16 Fuel Racks and Fuel Transfer Mechanical Equipment

471 I have undertaken a review of the mechanical equipment located within the spent fuel pool as part of my assessment, focussing on the fuel racks within the pool, and also the mechanical handling equipment for transferring fuel between the spent fuel pool and the reactor building pool during refuelling operations.

472 I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment principle EDR.1 (Ref. 4) states 'Due account should be taken of the need for structures, systems and components important to safety to be designed to be inherently safe or to fail in a safe manner and potential failure modes should be identified, using a formal analysis where appropriate.'

4.16.1 Assessment

4.16.1.1 Spent Fuel Racks

473 I have questioned EDF and AREVA regarding the design of the spent fuel racks, specifically covering the following points:

- The provision of features for assisting with the placement of fuel within the rack.
- The provision of features to assist remote handling, to ensure correct location of the fuel within the designated location.
- Any features within the design to minimise the impact loading due to a dropped load.
- The pedigree of the fuel rack design, including any relevant Operational Experience Feedback.

474 The storage equipment comprises a number of modules, each containing a number of individual cells, for the storage of both new and spent fuel, underwater within the spent fuel pool. Each module is a stainless steel structure, with the individual cells made of borated stainless steel to provide neutron absorption capability. The modules are supported from the bottom of the spent fuel pool using a ball pivot arrangement, with allowances for vertical adjustment, to ensure an even load distribution as dictated by civil engineering requirements.

475 During insertion and withdrawal of Fuel Assemblies (FA), they are protected by guide funnels with bevelled edges, and the surface finish within the cells is also controlled to prevent damage to the FAs. The FAs are also inserted and withdrawn vertically using the Spent Fuel Mast Bridge (SFMB), which is specifically designed to provide this purely vertical operation. The FA lowering lifting speed is also controlled as part of this operation.

476 The underwater fuel handling operations are operator led. The visibility of operations is assisted by CCTV cameras mounted on the SFMB, or anchored on the civil works. The position and movement of the SFMB in relation to the individual cells, and the occupancy of the cells, is also controlled through C&I systems, which I have not reviewed as part of my Mechanical Engineering assessment. However, I have noted that the SFMB can also be directly controlled by the operator, based on information presented by the C&I systems (controlling the FA locations).

477 EDF and AREVA have stated that any potential for damage to FAs within the individual cells due to dropped loads is minimised since the top of the FA is circa 200mm lower than the top of the cell. They have also stated that operating procedures forbid heavy load handling above the spent fuel pool. I have pursued this area separately in terms of my assessment of cranes, and specifically load path / route selection based on ALARP principles. I also note that the subject of dropped loads has been the subject of a Regulatory Observation, RO-UKEPR-070 (Ref. 11) by my Internal Hazards colleague.

478 EDF and AREVA have stated that Operational Experience Feedback is a continual process for the French fleet of NPPs, and as an example boronated stainless steel is used as a neutron absorbing material within the spent fuel pool, since their experience with Boral, (mixture of boron and aluminium), has identified some problems with blistering of the surface.

479 In summary, I am satisfied with the explanations provided by EDF and AREVA covering the Mechanical Engineering design of the fuel racks against SAP EDR.1.

4.16.1.2 Fuel Transfer Mechanical Equipment

480 I have questioned EDF and AREVA in respect of the mechanical equipment for transferring new and spent fuel between the spent fuel pool area, and the reactor building, during outages.

481 The equipment comprises a transfer tube, where fuel is moved using a conveyor trolley chain drive mechanism, and at each end a swinging chassis arrangement for reorientating the FA from the horizontal to the vertical, and vice versa. The FAs are transferred into and out of the arrangement by the Spent Fuel Mast Bridge, and Refuelling Machine, at each end respectively.

482 Interlocks ensure that the swinging chassis movements are coordinated with the horizontal movement of the conveyor trolley. The Fuel Transfer Facility is Seismic Class 1, and EDF and AREVA have also described the provision for recovery action in the event of credible faults, as follows:

- Redundant cables are provided for the winching devices, and in the event of rope breakage the remaining cable is sufficient to recover the situation.
- In the event of lifting equipment winch failure, (electrical or mechanical), the operator can use an emergency (manually operated) handwheel to recover the situation.

- In the event of travelling equipment failure, (electrical or mechanical), the operator can use a manual emergency travelling device to recover the situation.

483 EDF and AREVA have stated that this equipment and design is essentially the same as that used within the fleet of French NPPs, without any adverse Operational Experience Feedback reported. However, improvements have been made to the equipment, and EDF and AREVA have specifically stated that sensor redundancy has now been incorporated into the design.

484 EDF and AREVA have provided a good response to my enquiries, and have described equipment which is standard to the French fleet of NPPs. I have identified no specific concerns in this area from a Mechanical Engineering perspective, and I am satisfied with the designs as described against SAP EDR.1.

4.16.2 Findings

485 I have not identified any findings covering this area.

4.17 Radiation Waste Containers

486 At the start of my Step 4 assessment process I considered that there may have been some limited effort required in respect of Radiation Waste containers used within the Nuclear Power Plant.

4.17.1 Assessment

487 I have not identified any issues of a Mechanical Engineering nature with this Step 4 assessment worthy of consideration from a GDA perspective, and have liaised with the assessment discipline covering Waste and Decommissioning in coming to this conclusion.

4.17.2 Findings

488 I have not identified any findings covering this area.

4.18 Transportation Flasks

489 At the start of my Step 4 assessment process I considered that there may have been some limited effort required in respect of Transportation Flasks used within the Nuclear Power Plant.

4.18.1 Assessment

490 I have undertake a limited assessment of the transportation flasks (casks) associated with the spent fuel cask transfer facility, noting that when it is connected to the cask loading pit it becomes part of the spent fuel pool containment system; this is covered elsewhere in this report. I do not consider that there are any features of the UK EPR which specifically constrain the development of transportation flasks for the UK, recognising that it will be a number of years from initial criticality before fuel is removed from the spent fuel pool.

4.18.2 Findings

491 I have not identified any findings covering this area.

4.19 Mechanical Process Filters and IRWST Filtration System

492 Mechanical process filters can have a significant safety function in terms of the effective performance of a system, by removal of detrimental debris within the fluid, and as a consequence I undertook to review the following aspects in relation to the UK EPR submission:

- Purpose and safety functional requirements.
- Safety categorisation and classification.
- Examination, maintenance, inspection and testing regime.
- Operational Experience Feedback.

493 I have also assessed the IRWST filtration system design and qualification, since this plays an important part in the operation of the Safety Injection System, and the Containment Heat Removal System, to ensure the flow of water is not impeded by the build up of debris following a postulated LOCA. I consider the following Safety Assessment Principles to be relevant to this aspect:

- Safety Assessment Principle EDR.1 (Ref. 4) states 'Due account should be taken of the need for structures, systems and components important to safety to be designed to be inherently safe or to fail safe in a safe manner and potential failure modes should be identified, using a formal analysis where appropriate'.
- Safety Assessment Principle EMT.1 (Ref. 4) states 'Safety requirements for in-service testing, inspection and other maintenance procedures and frequencies should be identified in the safety case.'

4.19.1 Assessment

4.19.1.1 Mechanical Process Filters

494 Several systems within the UK EPR contain filters with their prime duty to manage the level of suspended solids within the fluids. Examples of such suspended solids include: oxidised materials, wear particles and resin fines downstream of the demineralisers.

495 Silica based filters are not used on any aqueous systems that contain enriched boric acid. EDF and AREVA explained that the boric acid dissolves the silica, which becomes extremely difficult to separate and recover from the process system downstream.

496 The UK EPR plant design has five different designs of filter, and the criteria for selecting a filter for a particular duty takes into account the following parameters:

- Radioactivity levels.
- Filtration requirements.
- Flow rate requirements.
- Plant spatial and interfacing constraints.

497 Of the five filter designs, four are of a vendor proprietary design and are procured on this basis. They are utilised in systems where the radioactivity levels are low, consequently disposal follows the low level waste disposal route.

498 Each filter is fitted with a differential manometer, which allows surveillance to be carried out on a routine basis to identify if the filter is operating within its design limits and is not blocked.

499 Ensuring the filter cartridge remains within the limits of low level waste allows the filter to be manually changed with the aid of mechanical lifting equipment, (in the cases when the mass is outside manual handling limits).

500 The disposal route for the Low Level Waste (LLW) filters is similar to that of existing French NPPs. Once a filter cartridge is removed from the plant it is placed into a drum. The drum is then transferred to a dedicated facility where the filter cartridge is then compacted to minimise the volume of waste generated. The drum is then transferred to a facility for suitable storage.

501 The “A” type filter is a specific design for incorporation within a higher radioactivity system. The design for the UK EPR is of the established type that is utilised within the existing French NPPs, which has benefitted from operational experience from the German NPPs. An example of this is the change in the filter design due to the interface with the filter change machine.

502 Where radiation dose levels dictate, filter cartridge handling and filter cartridge disposal are carried out by remote handling systems that incorporate biological steel shielding.

503 In addition to each filter being fitted with a manometer, filters located within high radioactive systems are also fitted with a permanent dose recorder.

504 Discussions highlighted that the design has evolved to group the high dose filters together to assist the interface with the filter change machine, and to enable a more efficient monitoring of the filter cartridge dose rates.

505 The filter design requirements such as filtration rate value, efficiency, and retention capacity are specified to be in accordance with European Standard NF EN 13443-2. Filter bodies and cartridge material is in accordance with NF EN 10204, which limits the cobalt content to no more than 0.2% if the material is in contact with the reactor system coolant.

506 EDF and AREVA described the sequential steps involved in replacing a low level radioactive filter cartridge. They highlighted that the draining of the filter is via a permanent pipework to the nuclear vent and drain system.

507 In response to questions EDF and AREVA advised that filter cartridges are replaced on the following three criterions:

- Pressure drop - Once a pressure drop (1- 2.5 bar filter dependant) is achieved, the filter cartridge is replaced; each filter cartridge is designed and tested to 6 bar.
- Radioactivity level - Once a certain dose rate is measured (based on the transport limit) the filter cartridge is replaced.
- Life time. - Once a filter cartridge has been in service for 5 years, the filter cartridge is replaced.

508 In response to further questioning, they advised the filter cartridge material is dependant on the system temperature and is either polypropylene or stainless steel. They also advised the filter cartridges carry a unique identification number but do not incorporate any specific poka-yoke features, so replacement is reliant on the operator reading and following the maintenance instructions. Although I consider such poka-yoke features to be useful in this application, I consider that the hazard of inadvertent use of an incorrect filter can also be satisfactorily accommodated by appropriately rigorous maintenance instructions, which I consider to be an Assessment Finding (**AF-UKEPR-ME-31**).

509 The safety classified steam generator blow down system has two sets of A1 type filters. One set (2 x 100%) is positioned upstream of the demineralisers, which limit suspended solids arriving from the steam generators. The other set (2 x 50%) is positioned downstream to restrict any suspended particles (resin fines) leaving the demineralisers.

510 The upstream filters' role is to clean the incoming water by means of a mechanical filtration. The two filters are installed in parallel, with one filtering 100% of the blowdown flow at any one time. The filtration is carried out by a multi-cartridge filter element.

511 Downstream of each of the demineralisers a 50% cartridge filter is positioned. Each filter captures resin particles in the event of a demineraliser strainer breaking or the resin beads being degraded. The filtration is carried out by a multi-cartridge filter element.

512 In summary from a Mechanical Engineering perspective I am satisfied with the design principles for the UK EPR mechanical process filters, and the remote handling of the higher level radioactivity filters within the UK EPR design against SAPs EDR.1 and EMT.1.

4.19.1.2 IRWST Filtration System

513 The In-containment Refuelling Water Storage Tank (IRWST) provides the head of water to supply the Safety Injection System comprising the Medium Head Safety Injection (MHSI) pumps and the Low Head Safety Injection (LHSI) pumps. Water is also supplied to the Containment Heat Removal System (CHRS) pumps from the IRWST. The long term provision of water to these pumps is achieved by recirculation of water into the IRWST from containment, which is filtered via Trash Racks protecting the floor drains, and retention baskets within the IRWST to capture and retain any LOCA induced debris, (TQ-EPR-533, Ref. 10).

514 The outflow from the IRWST is filtered via six dedicated strainers, each fitted with a backflushing system, to provide water to the MHSI, LHSI, and CHRS pumps. Each strainer comprises a wire mesh modular construction, with a surface area of 110 square metres. A backflushing system is also provided which can be initiated if the maximum pressure drop across the strainers is reached.

515 EDF and AREVA have described the qualification test programme for the filtration system design (Ref. 31), which covers the following technical parameters:

- Debris retention performance of the IRWST baskets.
- Filtration performance of the strainers.
- Backflushing functionality and performance for the strainers.

516 EDF and AREVA have stated that the qualification tests for the design are not complete, but preliminary results are satisfactory, and to date all validation criteria have been

achieved. For example the combination of retention baskets and strainers leads to a head loss across the strainer of 20 mbar, compared to the limit of 280 mbar at 30 degrees C; (this value guarantees Net Positive Suction Head performance of the MHSI, LHSI and CHRS pumps). Furthermore the operation of backflushing is stated as efficient, and is able to detach debris from the strainer, where it then settles on the IRWST floor.

517 I have reviewed the qualification test program document and consider it represents a rational and suitable process to verify the design performance of the IRWST mechanical filtration system. Given the reported status of the tests, I consider it to be an Assessment Finding (**AF-UKEPR-ME-32**) that the tests are satisfactorily completed to qualify the performance of the UK EPR design.

4.19.2 Findings

AF-UKEPR-ME-31: *The licensee shall make and implements adequate EMIT instructions to control the hazard of inadvertent use of an incorrect filter cartridge in a mechanical process filter. Target Milestone – fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-32: *The licensee shall ensure that the IRWST filtration system tests are satisfactorily completed to qualify the performance of the UK EPR design. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

4.20 Containment Doors and Hatches

518 I have assessed the Equipment Hatch and personnel access doors to the main containment, from a Mechanical Engineering perspective, since these provide the important safety function of containment of radioactive substances. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment principle ECV.1 (Ref. 4) states ‘Radioactive substances should be contained and the generation of radioactive waste through the spread of contamination by leakage should be prevented.’

4.20.1 Assessment

519 EDF and AREVA provided information covering the engineering design principles and features of the main containment Equipment Hatch, TQ-EPR-959 (Ref. 10). The purpose of the Equipment Hatch is summarised as follows:

- Introduction of equipment during erection (RPV, RCPs, main vessels etc).
- Introduction of equipment during outages (Multi Stud Tensioning Machine, spares).
- Third barrier of containment (against fission product release).

520 The equipment hatch engineering features include the hatch cover, flange clamping device, double seal to ensure leak tightness, and hatch opening and closing mechanism. In response to questions, EDF and AREVA stated that the flange double seal has a ten

year design life. EDF and AREVA also stated the categorisation / classification associated with these engineering features (Category A, Class 1), which is in line with my expectations from a Mechanical Engineering perspective.

521 EDF and AREVA explained the analyses carried out regarding the hatch and clamping device, which took account of a seismic event, natural settlement of concrete over the 60 year design life of the facility, severe accident conditions, and flange movements due to loading. In respect of the double seal arrangement EDF and AREVA stated that this is the same form of seal as qualified for the N4 series of plants. This EPDM material has satisfied the qualification requirements accounting for thermal ageing, accumulated radiation exposure, and thermodynamic conditions against the leak tightness requirements.

522 The inter-space within the double seal arrangement is ventilated to the annulus, which is held at a lower pressure than containment, and so any leakage past the first part of the seal will be extracted to this preferred route.

523 In respect of the flange joint, visual inspection of the seals is possible when the hatch is in the open position, and there is mandatory testing of the seal before start up. The seal is also required to be replaced every ten years. The six flange clamps also have sensors which confirm that the system is correctly aligned.

524 EDF and AREVA explained that the design of the equipment has benefitted from Operational Experience Feedback. Specifically ovalisation problems had not been considered in previous designs, and this phenomena had now been accommodated by accounting for the 60 year design life settlement of concrete, welding the shell following containment pre-stressing (to account for as built dimensions), and increasing the thickness of the flange to enable machining if distortion is larger than expected to achieve a suitable fit. Suitable heat treatment is also carried out on the hatch cover to alleviate distortion effects due to welding.

525 The hatch cover has its own integral lifting device incorporated within the design, allowing operation of the hatch to be independent of the Polar Crane and thus providing maintenance benefits. EDF and AREVA also stated that in the event that this lifting device fails, the Polar Crane could be used to lift the hatch, and furthermore the hatch could be lifted using manual effort via the hydraulic system (depending upon the nature of the failure). The hatch cover shell is also captured by the lifting rails, and EDF and AREVA stated that in the event of a credible lifting failure, the hatch cover would stay in position.

526 Overall I am satisfied with the justification of the Equipment Hatch, which is of an established design principle, and which has benefitted from Operational Experience Feedback from other similar applications. I am therefore content from a Mechanical Engineering perspective with this aspect of the design, in respect of GDA against SAP ECV.1.

527 EDF and AREVA also described the two Personal Access Hatches which allow the passage of operators and small equipment in and out of the Reactor Building. Both hatches are identical in terms of design, function and associated operational procedures, and they form part of the third barrier containment. The Personal Access Hatches comprise a large cylindrical sleeve, with two airlock doors, each of which is double lipped sealed using latest technology. EDF and AREVA described the categorisation / classification associated with the seals which was the same as for the Equipment Hatch (cat A, class 1), and also in line with my expectations. The qualification of the door seals

is also identical to that for the equipment hatch, with the addition of a mechanical aging test covering 4000 cycles, and a hybrid EPDM / silicon compound has been selected for the application, with a ten year design life. Manually operated hand-wheels are employed both inside and outside of each door to facilitate opening, and to alleviate the risk of personnel becoming trapped within the cylindrical airlock.

528 Overall I am satisfied with the justification of the Personnel Equipment Hatches, which are of an established design principle. I am therefore content from a Mechanical Engineering perspective with this aspect of the design, in respect of GDA against SAP ECV.1.

4.20.2 Findings

529 I have not identified any findings covering this area.

4.21 RPV Leak Detection System

530 The reactor pressure vessel head seal arrangement has the important safety function of containing the primary circuit fluid, and consequently the UK EPR reactor pressure vessel leak detection system is an important mechanical design arrangement to ensure that leakage is identified, monitored, and effective action is taken as necessary in line with safety case parameters. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment Principle ESS.3 states ‘Adequate provisions should be made to enable the monitoring of the plant state in relation to safety and to enable the taking of any necessary safety actions’.

531 As a consequence I targeted my assessment on the following aspects:

- Purpose, role and safety function.
- Evidence of equipment categorisation and classification.
- Examination, maintenance, inspection and testing regime.
- Consideration of relevant Operational Experience Feedback.

4.21.1 Assessment

532 EDF and AREVA described the UK EPR reactor vessel leak detection system, TQ-EPR-1276 (Ref. 10). The system design is described as follows:

- A double seal forms the primary containment barrier joining the reactor pressure vessel and the vessel head.
- Leakage from the inner seal is collected in a single length of pipework, that is fitted with instrumentation and is routed into the vent and drain system.
- A temperature sensor is fitted to the pipework, which provides an indication when the inner seal is passing.
- A pressure sensor is also fitted to the pipework, which provides an indication when the outer seal is passing (indicated by a decrease in pressure).

- A normally open motorised valve is located downstream of the two instruments.

533 During normal operations the reactor pressure vessel double seal arrangement provides the necessary containment barrier joining the reactor pressure vessel body and the reactor pressure vessel head. If the inner seal begins to pass, the temperature reading from the temperature sensor increases and provides a signal to close the motorised valve. At this point an alarm is activated in the main control room. Containment is maintained by the outer seal, which is now monitored by the pressure sensor. On the loss of the inner seal the leak detection pipework up to the motor operator valve develops the same pressure as the reactor vessel internals. The pressure sensor now provides the indication of the adequacy of the outer seal containment arrangement. A stabilised pressure reading indicates an adequate containment seal. A fall or fluctuation in the pressure reading indicates the seal arrangement is not adequate, the operator is informed by an alarm within the main control room and the reactor is shut down.

534 The reactor cooling system has a number of head tanks, which are the subject of daily accountancy checks to confirm the system leak rate. If the leak rate is found to be in excess of 230 l/hr the operating technical specification requires the reactor to be shut down.

535 The equipment safety categorisation / classification according to the latest methodology is as follows:

- The leak detection system up to and including the motor operated isolation valve has a safety Category C with a safety Class 3.
- The system has a seismic classification of SC2, as operability is not required following an earthquake.

536 I advised EDF and AREVA that the safety categorisation / classification of C3 is lower than my initial expectations, and I suggested that they review this area. Specifically I consider the isolation valve forms a principal means of ensuring a Category B safety function since on the loss of the inner seal, when the plant remains operational, the valve provides an important containment function. On this basis I expect the valve to have a safety classification 2 assigned to it, and I consider this to be an Assessment Finding (**AF-UKEPR-ME-33).**

537 EDF and AREVA advised that the proposed design has taken into account operational experience from the French N4 NPPs. The UK EPR proposed system design is of a simpler design than the existing N4 system. The N4 system incorporates a steam condenser and a water level measuring unit. The main disadvantage of this design is the need to condense the steam to enable a water level to be provided. Experience has shown that the operation of the steam condenser is also difficult; the system requirements to open and close valves are considered to have a detrimental effect on the seals' performance. Pressure fluctuations within the reactor coolant drain tank can also affect the water level within the measuring unit, which results in non accurate readings. Pressure fluctuations can also lead to spurious actuation of the system alarms.

538 In response to questions EDF and AREVA advised that the system design incorporates one temperature and one pressure sensor and that they are both of a recognised good design, being procured and qualified to specific nuclear codes and standards, which I consider is typical of Safety Class 2 equipment.

539 Other operational feedback focused on the seal detailed design. The external surface of the original seal included a 0.1 to 0.15mm silver coating. Periodic inspection identified

the reactor pressure flanges were the subject of corrosion and it was considered the silver coating was inadequate and was causing a complex electro chemical interaction between the RPV flange and the seal base material. The seal design was subsequently reviewed and by the early 1980s existing NPP seals were gradually being replaced by a Helicoflex™ seal, which is a toric seal with a spring rolled Inconel centre. The advantages of this design are that the silver coating has been increased to 0.3 mm, which has resulted in the RPV flange being less sensitive to corrosion, plus an increased tightness is achieved by the toric seal being a 'C' shape, which closes on compression. This is now the standard design of seal utilised across the French Fleet of NPPs with extensive operational experience demonstrating it to be adequate in achieving its design intent.

540 In response to further questions EDF and AREVA advised that in respect of the French fleet of NPPs:

- The seal is sacrificial and is replaced during each outage.
- Operational experience has identified that on occasions (during closing phases) the seals have lost their ability to maintain an adequate seal.

541 EDF and AREVA carried out a study to increase their understanding of this seal leakage phenomenon. The study identified the prime cause was inadequate cleanliness arrangements, indicating the key importance of adequately cleaning the interfacing surfaces prior to the fitting of the reactor pressure head. The investigations identified that the presence of foreign debris damaged the seals, and prevented them performing to their design intent.

542 Sizewell B NPP had a similar recorded closure leak in 2001, which on investigation was caused by the presence of debris, which resulted in damage to the seal and prevented it from performing to its design intent. Again a key lesson learned is the importance of implementing adequate cleanliness arrangements during an outage to ensure items such as seals can be satisfactorily fitted without being subjected to damage.

543 I have reviewed the Operating Experience Feedback relating to the Sizewell B NPP in respect of the RPV leak detection system, specifically the IAEA Incident Reporting System (IRS) report 7643, (Ref. 42).

544 In summary this IRS report describes an event which occurred in May 2001, whereby the RPV head seal leak detection system initially indicated leakage from the inner head O-ring seal. The inner seal leak detection path was then isolated and the reactor continued operation in line with operating instructions. Later in the fuel cycle airborne activity levels, humidity, and sump levels in containment provided evidence of leakage from the Reactor Coolant System (RCS), but this always remained well within the Technical Specification limit for unidentified leakage.

545 Despite several containment entries, the source of the RCS leak was not identified, and in particular the thermocouple on the outer seal leak detection system indicated no sign of leakage. Subsequently, and due to increased levels of leakage in containment, plans for the reactor shutdown were brought forward, and following a detailed leak search the source of the problem was identified as a leak from the RPV head outer O-ring seal. The subsequent investigation revealed that the outer O-ring leak detection system would not reliably detect outer O-ring leakage when the reactor is at operating conditions. This outer O-ring leak detection system comprised a single hole in the flange to collect any fluid, and in this instance the actual leak site was at the opposite side of the RPV.

Furthermore the high temperature of the RPV meant that any leakage would rapidly boil away, and not provide an indication of a leak by fluid collection by the outer O-ring leak detection system.

546 The IRS report concludes that although the inner O-ring leak detection is effective, the outer O-ring detection system has reduced reliability, and so if operating on the outer seal only, alternative indications such as containment activity, humidity, and drainage must also be used to detect outer seal failure.

547 The source of the leak itself was considered to be loose particle debris in the flange area which had occurred during maintenance activities, which is a known problem for RPV head closure sealing.

548 The consequence of a leak from the RPV is also boric acid crystal deposition, and associated corrosion of the low alloy steel of the RPV outer surfaces, which has been a significant problem for NPPs in the past.

549 In respect of the design of the RPV leak detection system described by EDF and AREVA, for the UK EPR leakage past the outer seal is deduced by a pressure fluctuation in the seal interspace, and as such the design does not rely on detecting any fluid that has passed beyond this second seal. In this respect the UK EPR design is not directly susceptible to the phenomenon described by this OEF. Nevertheless, I consider this should be supplemented by operational requirements to detect any passing of the outer seal, such as measurements of containment activity, humidity, and drainage, which I consider to be an Assessment Finding (**AF-UKEPR-ME-34**).

550 I further consider it to be an Assessment Finding (**AF-UKEPR-ME-35**) that a future licensee should develop adequate EMIT procedures for the detection of leaks of boric acid generally within containment.

551 Notwithstanding these Assessment Findings I am satisfied with the UK EPR design against SAP ESS.3.

4.21.2 Findings

AF-UKEPR-ME-33: *The licensee shall review the safety categorisation and classification of the RPV leak detection system to ensure it is adequate, since I consider that the declared Cat C / Class 3 assignment does not correctly reflect the importance of this system to safety. However, I recognise that this assignment needs to be consistent with the methodology adopted against the UK expectations in this area, in response to cross-cutting Regulatory Observation RO-UKEPR-043. Target Milestone – Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-34: *The licensee shall review the safety case Operational Limits and Conditions to ensure that procedures are adequate to detect any passing of the outer RPV seal, such as measurements of containment activity, humidity, and drainage. Target milestone – fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

AF-UKEPR-ME-35: *The licensee shall develop adequate EMIT procedures for the detection of leaks of boric acid generally within containment, against the background of worldwide Operational Experience Feedback. Target Milestone – fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.*

4.22 Nuclear Vent and Drains

552 I consider the containment of radioactive substances, prevention of leakage, and limitation of the spread of any contamination to be important safety functions for a NPP. On this basis I pursued a line of enquiry by targeting the EDF and AREVA valve leak collection and detection arrangements. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment principle ECV.1 (Ref. 4) states ‘Radioactive substances should be contained and the generation of radioactive waste through the spread of contamination by leakage should be prevented.’

4.22.1 Assessment

553 EDF and AREVA described the UK EPR valve leak recovery system. The information identified that a valve leak off recovery system is incorporated within the valve design and connected to the Nuclear Vent and Drain System (NVDS) when all of the following conditions are met:

- System fluid is radioactive or the valve forms the containment isolation.
- System pipework diameter is greater than 50mm.
- Valve is equipped with a packing box.

554 A leak off recovery system typically consists of a flexible connection from the valve body, which manages the valve and pipework loading requirements. The flexible connection is attached to a pipeline containing a sight glass, which is routed to the nuclear vent and drain system where the process fluid is collected in tanks, which have level indication incorporated.

555 Through discussion EDF and AREVA explained that several leak recovery lines connect to a particular collection tank. I consider during operations, this strategy would make it more difficult to identify the source of a leak, but accept that the provision of extensive separate systems is not reasonably practicable in general.

556 EDF and AREVA provided further information on the NVDS that covered its role, safety requirements, categorisation of effluents, and the various effluent routes. The description covered the primary effluents (from the primary coolant), process drains, chemical drains and the floor drains, which are sub-divided into three categories, based on their level of contamination. They described the routing of the various effluent streams, and some associated testing to further sentence the effluent to the correct routing, which is based on the level of contamination.

557 EDF and AREVA clarified that the NVDS sump level measurement principle is performed by two level detectors. The measurement is based on the management of 4 alarms, 2 alarms for the low level (low and low-low) and 2 alarms for the high level (high and high-high), with each sensor assigned to a high and a low alarm role for diversity. The type of

sensor used is dependant on the functional requirements, and in this case the design choice is based on level switches with no requirement for continuous monitoring. The adopted technology is dependant on the size of the sump, layout constraints and the operating conditions for the sensor during normal and accident conditions. Two types of level sensors can be utilised to perform this duty either an 'on-off' or an analogue sensor.

558 Analogue sensors used for the NVDS sump level measurements use either of the following principles:

- Ultrasound.
- Radar wave.
- Capacity probe.

559 'On-off' sensors used for the NVDS sump level measurements use either of the following principles:

- Floats.
- Masses with magnetic transmission.

560 A dedicated pump is automatically started on the level sensor indicating a high level, which is automatically stopped at the level sensor indicating a low level. The low-low signal is provided for pump protection on the loss or malfunction of the low signal. The high-high signal initiates an alarm in the main control room, and the operator follows a suitable operating instruction.

561 EDF and AREVA clarified that all the sumps include the requirement to protect the pumps against debris by the inclusion of a strainer. EDF and AREVA stated this requirement is captured within the equipment technical procurement specification.

562 In response to questions, EDF and AREVA:

- Agreed to provide more information in respect of the arrangements if there was insufficient time to undertake a full analysis of effluent, in terms of its routing and associated safeguards. Furthermore, they also agreed to confirm that for the floor drains, there were no circumstances where effluent could be directed to a lower category of drain than required by its level of contamination.
- Stated that sump sizes were based on estimated levels of effluents, and experience from the French operating NPPs. EDF and AREVA also confirmed that the sumps are wet sumps, having a minimum level of fluid associated with the pumping requirements.
- Stated that improvements have been made to the NVDS design over previous reactors to enable boron recycling, and the separation of the floor drains into three streams.
- Provided further clarification and an explanation in respect of the use of gulleys in building rooms, specifically stating that these are provided if a sprinkler is located within a room, and also for maintenance where a water intake is provided.

563 Through further discussion on the sentencing of liquids through the various treatment options, EDF and AREVA explained that the decision process was affected by both the radioactive contamination potential within the fluid, and the potential chemical content of the fluid.

564 In response to questions I was still unclear regarding the EDF and AREVA rationale for routing effluent to a particular drain if a sump is found to be overflowing and the effluent analysis is not complete.

565 I decided to discuss this subject with the Waste Management and Decommissioning discipline, since I considered that the matters under consideration now fell outside the scope of Mechanical Engineering, and I was not able to judge the significance of the issues described. I have now agreed for this subject to be transferred to the Waste Management and Decommissioning discipline for their consideration.

566 In summary, I have not identified any Mechanical Engineering concerns from a GDA perspective, and am therefore satisfied with the designs described against SAP ECV.1.

4.22.2 Findings

567 I have not identified any findings covering this area.

4.23 Main Feed Water System

568 I have reviewed the steam generator Main Feed Water System during my Step 4 assessment, since this system has the safety function of providing cooling to the primary circuit under both normal and fault conditions, in addition to reactivity control and containment of radioactive substances. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment Principle EHT.1 states ‘Heat transport systems should be designed so that heat can be removed or added as required.’

4.23.1 Assessment

569 EDF and AREVA provided a description of the Main Feed Water System (MFWS) which has a duty function in normal operations to supply conditioned water to the four steam generators during start-up, shut-down, and for power operations. The MFWS is also required to act in certain accident conditions to reach the appropriate controlled state, again by supplying water to the steam generators.

570 The MFWS comprises four trains located in the four safeguard Buildings, fed from a common header located in the turbine island. The valve arrangements associated with the general system design of the MFWS comprise control valves (Very Low Load, Low Load and Full Load), plus isolation valves and check valves to provide system isolation.

571 The safety functions associated with the MFWS are as follows:

- Reactivity Control – to prevent overcooling of the primary system (moderation increase).
- Heat transfer and removal – during normal operation and following reactor trip.
- Containment – containment of radioactivity following Steam Generator Tube Rupture.

572 EDF and AREVA described the safety categorisation / classification associated with the MFWS to reflect the above safety functions, which was in line with my expectations from a Mechanical Engineering perspective.

573 In responses to further questions EDF and AREVA explained that each Full Load Isolation Valve (FLIV) incorporates a limit switch that sends a signal to the Main Control Room on the isolation of a line. Each valve is tested during the commissioning phase and is the subject of periodic testing during the NPP lifetime.

574 The site commissioning tests confirm each FLIV functionality and isolation requirement from both a standalone perspective and a full integrated system perspective.

575 EDF and AREVA stated that during the NPP lifetime the following periodic tests are to be performed:

- During each outage a valve closure test is undertaken, including an evaluation of the time taken to close the valve. Each valve is tested twice since the test is done for each manifold.
- A second test is performed every 2 months and consists of testing the availability of the FLIV manifolds.

576 EDF and AREVA described the maintenance associated with the MFWS, noting that preventative maintenance during normal power operation is not considered for the MFWS, but that safety classified equipment can be visually inspected outside containment. Inspection of the MFWS inside containment can only be undertaken during shut-down states and in a limited timescale. For these reasons the amount of welding required to be inspected is low, and exterior inspections are preferred to limit radiation exposure of personnel. During shut-down periodic tests are undertaken to verify safety functions, specifically focused on isolation valve operability.

577 My assessment of the System Design Manual, Main Feedwater System – Part 3: System Design (Ref. 43) has confirmed the Main Isolation Valve is specified to the system bounding pressure rating of 150 bar abs, which is in line with my expectations.

578 In summary, I am satisfied with the system design of the MFWS from a Mechanical Engineering perspective against SAP EHT.1.

4.23.2 Findings

579 I have not identified any findings covering this area.

4.24 Emergency Feed Water System

580 I have reviewed the steam generator Emergency Feed Water System during my Step 4 assessment, since this system has the safety function of providing cooling to the primary circuit under fault conditions, in addition to reactivity control and containment of radioactive substances. I consider the following Safety Assessment Principle to be relevant to this aspect:

- Safety Assessment Principle EHT.1 states 'Heat transport systems should be designed so that heat can be removed or added as required.'

4.24.1 Assessment

581 The Emergency Feed Water System (EFWS) comprises one dedicated line for each of the four steam generators, located in each of the Safeguard Buildings leading into the

reactor building. Each train comprises a separate water storage tank, a pump, plus control, isolation, and check valves. The piping layout allows interconnection between the four lines of the EFWS, controlled via manually operated isolation valves. The EFWS is supplied via the Demineralised Water Distribution System, and also interfaces with the Chemical Injection System which is used as appropriate during outages for injection of chemicals into the primary circuit. The Fire Water Storage System can also be used to refill the EFWS tanks as necessary in case of Loss of Ultimate Heat Sink.

582 The EFWS is not designed to be used during normal operation, but is used to supply feedwater to the steam generators in the event of failure of the MFWS systems. The safety functions associated with the EFWS are described as follows:

- Reactivity Control – in the event of a main steam line break, the affected steam generator is isolated to prevent excessive primary circuit cooldown, and is fed by the EFWS (with pressure relief provided by the Main Steam Relief Train via the PORV).
- Residual Heat Removal – to allow cooling of the primary circuit by the steam generators under specified fault conditions.
- Containment – to allow isolation of the affected steam generator in the event of a Steam Generator Tube Rupture, which is fed by the EFWS, (with pressure relief provided by the Main Steam Relief Train via the PORV).

583 The EFWS trains are physically separated within the four safeguard buildings. All trains are backed by the Main Diesel Generators, and two of the trains are also backed up by the Ultimate Diesel Generators. Each pump is sized to accommodate 50% of the maximum total required flow.

584 The safety functions associated with the EFWS are controlled by periodic tests, comprising pump tests, injection tests, re-alignment tests, tank replenishment tests, and valve tests.

585 One of the EFWS safety functions is also to provide an automatic SG level adjustment in restoring a safe shutdown state when operating instructions require the SG levels to be managed via the EFWS flow control valve (ASGi310VD). The throttle valve is also required to control the inherent dilution facility.

586 EDF and AREVA have also clarified that isolation of the EFWS train is also required under certain fault conditions (TQ-EPR-1245, Ref. 10) to:

- Limit the filling of the SGs, in the event of a 1 or 2 tube SGTR break to limit an increase in water inventory from the supply of the reactor coolant.
- Limit consumption of the EFWS tanks in feed water line break or steam line break conditions.
- Limit the containment pressure and temperature in a feed water line break event.
- Ensure core sub-criticality in a controlled state in the event of a steam line break.

587 This system isolation is provided by the design incorporating two valves, a check valve (ASGi411VD) and an automatic isolation valve (ASGi410VD). An additional control valve (ASGi310VD) is incorporated into the system to control and manage fluid flow rate.

588 EDF and AREVA's fault studies analysis requires the isolation to be achieved within 60s. They stated the isolation requirements are tested during initial plant start up and during subsequent EMIT activities.

589 The four EFWS trains are connected to a discharge header that allows the system to inject via any of the train pumps. In addition, the suction header allows the use of water from any of the four Emergency Feed Water Storage tanks.

590 A pump discharge re-alignment is required in the event of a feed water line break or a SGTR, to ensure an adequate level is achieved within the SGs, plus during a station blackout in a State "A" scenario.

591 EDF and AREVA explained that their fault studies analysis indicates that this pump re-alignment, which is achieved by manual adjustment of a number of valves, is required to be undertaken on demand within 1 hour. The time period of 1 hour drives the valve design to be of a manual operation in preference to a motorised valve type. In addition a study was undertaken during the basic design phase, which reviewed and evaluated the potential benefits of using a motorised valve in preference to a manual operated type. The PSA analysis showed that a motorised valve operated from the control room is marginally more reliable than a local operated manual valve. Because of only marginal benefits, EDF and AREVA consider the chosen design to be acceptable and ALARP. I accept this explanation from a Mechanical Engineering perspective.

592 In summary, I am satisfied with the system design of the EFWS from a Mechanical Engineering perspective against SAP EHT.1.

4.24.2 Findings

593 I have not identified any findings covering this area.

4.25 Overseas Regulatory Interface

594 In accordance with its strategy, HSE collaborates with Overseas Regulators, both bilaterally and multinationally.

595 Bilateral collaboration:

- HSE's Nuclear Directorate (ND) has formal information exchange arrangements to facilitate greater international co-operation with the nuclear safety regulators in a number of key countries with civil nuclear power programmes. These include the:
    - i) US Nuclear Regulatory Commission (US NRC).
    - ii) French Nuclear Regulator (ASN).
    - iii) Finnish Regulator (STUK).

596 Multilateral collaboration:

- ND collaborates through the work of the International Atomic Energy Agency (IAEA) and the Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (OECD-NEA). ND also represents the UK in the Multinational Design Evaluation Programme (MDEP) - a multinational initiative taken by national safety authorities to develop innovative approaches to leverage the resources and knowledge of the national regulatory authorities tasked with the review of new reactor power plant designs. This helps to promote consistent nuclear safety assessment standards among different countries.

597 I have had some discussions with the US NRC as part of my assessment process, since they are undertaking a similar exercise in respect of the EPR NPP for licensing in the United States of America. This exercise has provided a useful exchange of information and has helped to guide my assessment in certain areas, the outcome of which is reported within the text of this document.

4.26 Interface with Other Regulators

598 I have worked with the Environment Agency as an integral part of the assessment process, although for the EDF and AREVA UK EPR design I have not identified any specific areas of Mechanical Engineering interest where detailed liaison has been considered necessary.

4.27 Other Health and Safety Legislation

599 I have considered conventional safety legislation in a general sense as part of my assessment process, although I have not undertaken a systematic review in this respect, since I do not consider it appropriate for this scope and level of assessment. I have focused my attention on the nuclear hazard, in line with the HSE-ND mission, to protect people and society from the hazards of the nuclear industry. Through my interactions with EDF and AREVA, I have reminded them of the requirement for any UK EPR constructed in the UK to comply with all relevant health and safety legislation, i.e. the Health and Safety at Work etc Act 1974 and its relevant statutory provisions.

5 CONCLUSIONS

600 This report presents the findings of the Step 4 Mechanical Engineering assessment of the EDF and AREVA UK EPR reactor.

601 The Step 4 assessment in my topic area commenced with consideration of the relevant chapter (s) of the PCSR and supporting references available at that time, and these are referred to as appropriate in this report. As the GDA submission developed during Step 4, in response to my regulatory questions, amendments were made as appropriate to the PCSR and its supporting references. A review has been made of the updates to the GDA submission in my technical topic area and the conclusion of this review is that:

- The updates to the GDA submission are not fully as expected, and some further amendments / justification to the consolidated PCSR and / or supporting references will be required. These will be progressed through GDA Issue **GI-UKEPR-CC-02** (Ref. 47). However, these actions do not have a significant impact on my assessment report and in my technical topic area. The consolidated PCSR (Ref. 46) and its supporting references are therefore acceptable as the reference point for an Interim Design Acceptance Confirmation.

602 To conclude, I am satisfied with the claims, arguments and evidence laid down within the PCSR (Ref. 46) and supporting documentation for Mechanical Engineering as listed in the Submission Master List (Ref. 12). I consider that from a Mechanical Engineering view point, the EDF and AREVA UK EPR design is suitable for construction in the UK. However, this conclusion is subject to assessment of additional information that becomes available as the GDA Design Reference is supplemented with additional details on a site-by-site basis.

5.1 Key Findings from the Step 4 Assessment

5.1.1 Assessment Findings

603 I conclude that the Assessment Findings listed in Annex 1 should be programmed during the forward programme of this reactor as normal regulatory business.

5.1.2 GDA Issues

604 I have not identified any GDA Issues associated with the Mechanical Engineering aspects of the UK EPR safety submission, through undertaking my assessment activity.

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Table 1

Relevant Safety Assessment Principles for Mechanical Engineering Considered During Step 4

SAP No.	SAP Title	Description
FP series	Fundamental principles	FP.1 to FP.8
SC series	Safety cases	SC.1 to SC.8
EKP series	Key principles	EKP.1 to EKP.5
ECS series	Safety classification and standards	ECS.1 to ECS.5
EQU series	Equipment qualification	EQU.1
EDR series	Design for reliability	EDR.1 to EDR.4
EMT series	Maintenance, inspection and testing	EMT.1 to EMT.8
EAD series	Aging and degradation	EAD.1 to EAD.5
ELO series	Layout	ELO.1 to ELO.4
EHA series	External and internal hazards	EHA.1 to EHA.17
EPS series	Pressure systems	EPS.1 to EPS.5
ESS series	Safety systems	ESS.1 to ESS.27
EES series	Essential services	EES.1 to EES.9
ECV series	Containment and ventilation	ECV.1 to ECV.10
EHT series	Heat transport systems	EHT.1 to EHT.5

Annex 1

Assessment Findings to be Addressed During the Forward Programme as Normal Regulatory Business

Mechanical Engineering – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-ME-01	The licensee shall make available evidence of the detailed design substantiation, Factory Acceptance Test (FAT) information, and Site Acceptance Test (SAT) information for individual mechanical items and their associated systems, which are important to safety.	Fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-02	The licensee shall ensure that all System Design Manuals (SDM) are reviewed and revised appropriately to align with the UK EPR safety categorisation and classification methodology, which is an accepted outcome of the work to resolve cross-cutting Regulatory Observation RO-UKEPR-043.	Mechanical, Electrical and C&I Safety Systems, Structures and Components -inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-03	The licensee shall generate appropriate evidence that Equipment Qualification is adequately specified for all mechanical items important to safety, accounting for new suppliers and the overall UK context.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.

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Assessment Findings to be Addressed During the Forward Programme as Normal Regulatory Business

Mechanical Engineering – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-ME-04	The licensee shall ensure that safety function categorisation and equipment classification is specified for all mechanical items important to safety, specifically including equipment which is the source of postulating initiating events (i.e. safety related systems, also termed duty systems). I consider that initially this exercise should focus on the major items of mechanical equipment, at an appropriate level to reflect the GDA workscope.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-05	The licensee shall ensure that the identification of plant limits and conditions, and EMIT requirements, from the safety case is completed to cover all Mechanical Engineering equipment important to safety. The licensee shall generate sufficient safety case information to satisfy the requirements of LC 23 and LC 28, and specifically a suitable interface shall be established to facilitate transfer of this information from the Responsible Designer, in due course.	Fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-06	The licensee shall review and consider alternative materials to Stellite™ for applications within the NPP domain, and generate evidence to ensure that material selection is ALARP for the UK EPR in respect of the use of Stellite™.	Mechanical, Electrical and C&I Safety Systems, Structures and Components – delivery to site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.

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Assessment Findings to be Addressed During the Forward Programme as Normal Regulatory Business

Mechanical Engineering – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-ME-07	The licensee shall ensure that the RCCA CMF-013 design change is fully substantiated and reflected in all design and safety documentation.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-08	The licensee shall generate evidence to demonstrate that the CRDMs meet their seismic design intent.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-09	The licensee shall generate the approved copy of the CRDM Endurance Test Report that records the CRDM test evidence.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.

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Assessment Findings to be Addressed During the Forward Programme as Normal Regulatory Business

Mechanical Engineering – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-ME-10	The design code identification of “n.a.” (assumed to mean “not applicable”) to the CRDM drive rod and the displacement limiter is not to my expectations. The licensee shall generate evidence that the CRDM and its constituent components are assigned with appropriate Mechanical Engineering design / material codes, which are commensurate to their importance to safety.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-11	The licensee shall clarify and justify the operating limits and conditions of the Reactor and the Reactor Coolant Pumps on the loss of the Chemical and Volume Control System seal injection system and / or the thermal barrier heat exchanger.	Fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-12	The licensee shall provide evidence to demonstrate the Reactor Coolant Pump maintenance activities meet applicable Conventional Safety Regulations.	Fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-13	The licensee shall ensure that the Reactor Coolant Pump maintenance requirements are adequately specified to meet the safety functional requirements throughout their operational life.	Fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.

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Assessment Findings to be Addressed During the Forward Programme as Normal Regulatory Business

Mechanical Engineering – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-ME-14	The licensee shall ensure the design of all rigging equipment associated with lifts of nuclear safety significance is completed, and in doing so shall systematically review these rigging arrangements to identify faults, and review and implement reasonably practicable improvements to either eliminate such faults by design, or limit their frequency by the provision of engineered protection systems.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-15	The licensee shall specify the design load paths for the RPV missile protection slabs based on ALARP principles, based on completion of any necessary dropped load consequence studies.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-16	The licensee shall specify the choice of RPV head lift load path / route based on the ALARP considerations described in the response to RO-UKEPR-052, UK EPR GDA – Management of Nuclear Safety Significant Lifting, ECEMA101802 Revision B.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.

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Assessment Findings to be Addressed During the Forward Programme as Normal Regulatory Business

Mechanical Engineering – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-ME-17	The licensee shall specify the choice of Fuel Pool Stop Gate load path / route based on the ALARP considerations described in the response to RO-UKEPR-052, UK EPR GDA – Management of Nuclear Safety Significant Lifting, ECEMA101802 Revision B.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-18	The licensee shall ensure that all lifts of nuclear safety significance are identified, and safe load paths are specified through appropriate design and safety documentation, and procedures.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-19	The licensee shall establish an appropriate filter change doctrine for all safety important filters within the nuclear ventilation systems.	Fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.

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Assessment Findings to be Addressed During the Forward Programme as Normal Regulatory Business

Mechanical Engineering – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-ME-20	The licensee shall verify the site specific design air temperatures and humidity values against those used as the basis for the UK EPR design, to ensure that the nuclear ventilation systems can adequately perform their safety functions.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - delivery to site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-21	The licensee shall ensure that the proposed modification to the nuclear ventilation system, described as CMF-020 (Confinement – Modification of Ventilation Systems) is fully incorporated into the UK EPR design and safety documentation.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-22	The licensee shall ensure that fume cupboards within the UK EPR are not used for the containment of radioactive substances.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.

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Assessment Findings to be Addressed During the Forward Programme as Normal Regulatory Business

Mechanical Engineering – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-ME-23	The licensee shall ensure that the analysis of a passive failure of the single branch connection of the Distribution of Demineralised Reactor Water System to the CCWS is undertaken, and any resulting findings are incorporated into all necessary design and safety documentation.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-24	The licensee shall assess the practicability of inspecting and / or replacing detrimentally affected sections of the CCWS in respect of corrosion, and implements any necessary ALARP improvements which are identified.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-25	The licensee shall quantify the heat transfer capability of the ultimate heat sink to operate in closed loop mode, specifically by the use of the UCWS diversification pipe, and develop any necessary operating instructions to provide a capability in this scenario.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.

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Assessment Findings to be Addressed During the Forward Programme as Normal Regulatory Business

Mechanical Engineering – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-ME-26	The licensee shall ensure that the safety class of the ultimate diesels is reviewed and justified, since I consider that the declared Class 3 assignment does not correctly reflect the importance of this equipment to safety. However, I recognise that this assignment needs to be consistent with the methodology adopted against the UK expectations in this area.	Install diesel generators complete as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-27	The licensee shall ensure that the diesel EMIT requirements are adequately transferred into the Plant Maintenance Schedule, (although the topic is also covered generically under cross-cutting Regulatory Observation RO-UKEPR-055).	Fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-28	The licensee shall ensure that the UK EPR diesel systems adequately take into account the regulation amendment in respect of fuels, (Motor Fuel (Composition and Content) Regulations 1999), in terms of meeting their safety functional requirements.	Install diesel generators complete as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-29	The licensee shall ensure that the Spent Fuel Cask Transfer Facility mechanical interface is adequate to ensure transfer of the spent fuel out from the NPP.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.

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Assessment Findings to be Addressed During the Forward Programme as Normal Regulatory Business

Mechanical Engineering – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-ME-30	The licensee shall ensure that the output from the Regulatory Observation from the Fault Studies discipline (RO-UKEPR-075), relating to the safety case justification covering the spent fuel pool, is reviewed to ensure that all mechanical items important to safety are covered by adequate safety function categorisation and classification, and systems and equipment are specified accordingly.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-31	The licensee shall make and implements adequate EMIT instructions to control the hazard of inadvertent use of an incorrect filter cartridge in a mechanical process filter.	Fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-32	The licensee shall ensure that the IRWST filtration system tests are satisfactorily completed to qualify the performance of the UK EPR design.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.

Annex 1

Assessment Findings to be Addressed During the Forward Programme as Normal Regulatory Business

Mechanical Engineering – UK EPR

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-UKEPR-ME-33	The licensee shall review the safety categorisation and classification of the RPV leak detection system to ensure it is adequate, since I consider that the declared Cat C / Class 3 assignment does not correctly reflect the importance of this system to safety. However, I recognise that this assignment needs to be consistent with the methodology adopted against the UK expectations in this area, in response to cross-cutting Regulatory Observation RO-UKEPR-043.	Mechanical, Electrical and C&I Safety Systems, Structures and Components - inactive commissioning as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-34	The licensee shall review the safety case Operational Limits and Conditions to ensure that procedures are adequate to detect any passing of the outer RPV seal, such as measurements of containment activity, humidity, and drainage.	Fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.
AF-UKEPR-ME-35	The licensee shall develop adequate EMIT procedures for the detection of leaks of boric acid generally within containment, against the background of worldwide Operational Experience Feedback.	Fuel on-site as this is the appropriate point when sufficient evidence should be available to demonstrate this requirement for Mechanical Engineering.

Note: It is the responsibility of the Licensees / Operators to have adequate arrangements to address the Assessment Findings. Future Licensees / Operators can adopt alternative means to those indicated in the findings which give an equivalent level of safety.

For Assessment Findings relevant to the operational phase of the reactor, the Licensees / Operators must adequately address the findings <u>during</u> the operational phase. For other Assessment Findings, it is the Regulators' expectation that the findings are adequately addressed no later than the milestones indicated above.

Annex 2

GDA Issues – Mechanical Engineering – UK EPR

There are no GDA Issues for this topic area.