Office for Nuclear Regulation

ENSREG Topical Peer Review on Ageing Management

United Kingdom National Assessment Report

December 2017

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Executive summary

The European Union's Nuclear Safety Directive requires the member states to organise a topical peer review (TPR) every six years with the first one starting in 2017. ENSREG has selected ageing management at nuclear power plants and research reactors as the topic for the first review. The first stage of the peer review is for each member state to produce a national assessment report. This is the UK National Assessment Report, which will be used in the later stages of the peer review.

WENRA has developed a technical specification for the national assessment reports to ensure that they are all produced to a common standard. This report has been produced in accordance with that specification which includes a well-defined content and identifies report sections to be written by licensees and by regulators. These sections have been amalgamated by ONR into a single UK national report which includes ONR's regulatory assessment and overall conclusions.

The scope of this report is the fifteen operating nuclear power reactors owned by EDF Nuclear Generation Limited (EDF-NG) on eight stations and the twin reactors currently under construction at Hinkley Point C owned by EDF New Nuclear Build (NNB). Whilst they are both part of the EDF group, the two organisations are separate companies and separate nuclear site licensees.

Both EDF-NG and NNB have ageing management programmes (AMP). EDF-NG has a full AMP across its sites, whereas NNB is in the early stages of developing its AMP and hence much of the detail has yet to be developed.

This report describes the overall AMP used by each company, which comprises the processes and systems used to manage ageing on the sites. Both companies have a management system which incorporates the elements needed for ageing management along with other requirements. The relevant parts of the ageing management system are described within the context of the international standards for ageing management. ONR has assessed the overall AMPs and concludes that they are adequate, given the lifecycle stage of the associated plants.

To illustrate the application of the overall AMP, the specification requires each member state to describe and assess the AMP for specific technical examples of structures, systems and components (SSC) prescribed by the specification and for the UK, for both licensees, these are:

- Electrical cabling
- Concealed pipework
- Reactor pressure vessels
- Concrete containment structures (Pressurised Water Reactors only)
- Pre-stressed concrete pressure vessels (EDF-NG Advanced Gas Cooled Reactors only)

Within each of these examples, the specification requires ageing management of specific components or types of component to be described and assessed.

ONR has in turn assessed the AMPs for each of the specific examples of SSCs and found them to be adequate given the lifecycle stage of the plants.

In addition to finding the overall AMP and AMPs for the example SSCs to be adequate, ONR's assessment has found that there are areas where improvement would be beneficial and proportionate. These have been clearly identified to the licensees and timescales agreed for these to be implemented.

Overall, ONR's assessment has found that both the operating reactors and the reactors under construction at Hinkley Point C have adequate ageing management programmes given the stages that each nuclear power plant is in its lifecycle. Some secondary beneficial improvements have been found and a programme for improvement for each licensee has been developed.

This report will be used as the basis for the next stage of the TPR, when it will be peer-reviewed by the other states participating in the TPR, after which the programmes for improvement for the two licensees will be finalised.

Table of contents

0 Preamble

0.1. The Council of the European Union’s (EU) Nuclear Safety Directive requires the member states of the EU to undertake a topical peer review (TPR) every six years with the first starting in 2017. The peer review process consists of three stages:

- Production of a national assessment report (NAR) on the selected topic by each member state;
- Written comments on member states’ NARs;
- A peer review workshop.

0.2. The topic that has been selected for the first review is ageing management of nuclear power plants and research reactors. This report is the national assessment report for the United Kingdom.

0.3. This report has been written to a standard technical specification (Ref 1) which applies to all states participating in the TPR. This is prescriptive, both in terms of the structure and the content of the national assessment reports. The objective of this is to ensure an effective peer review using consistent reports for all participants in the TPR.

0.4. The framework of the national assessment reports is based on Issue I, “Ageing Management”, of the Western Nuclear Regulators Association (WENRA) Safety Reference Levels (SRL) for Existing Reactors (Ref 2).

0.5. In accordance with the technical specification, ageing is considered as a process by which the physical characteristics of a structure, system or component (SSC) change with time (ageing) or use (wear-out). A related topic in determining the safety of nuclear installations is obsolescence of SSCs, i.e. their becoming out of date in comparison with current knowledge, standards and technology. Obsolescence is viewed as a different issue that is not related to the plant items within the scope of this TPR. Obsolescence is therefore not included within the scope of this and other states’ NARs.

0.6. Ageing management is understood as the engineering, operations and maintenance actions undertaken by a licensee to prevent or to control within acceptable limits, ageing degradation of the SSCs in its installation. With regard to safety, it ensures the availability of required safety functions throughout the service life of the plant, with account taken of changes that occur with time and use.

0.7. Ageing management of nuclear installations is applied to many SSCs. Those which are managed for economic purposes only are outside the scope of the TPR. The NAR will instead focus on SSCs important to nuclear safety, which includes:

- SSCs important to safety that are necessary to fulfil the fundamental safety functions;
- Other SSCs whose failure may prevent SSCs important to safety from fulfilling their intended functions.

Within the specification, the use of the term SSC should be taken to mean an SSC important to safety as defined above.

0.8. A key SRL within Issue I (Ref 2) is SRL I1.1, which requires each operating organisation to have an ageing management programme (AMP). This SRL further describes the AMP as:

> “*an integrated approach to identifying, analysing, monitoring and taking corrective actions and document the ageing degradation of structures, systems and components*“

Thus, the objective of AMPs with regard to safety is to ensure the prevention, the timely detection and mitigation of any age related degradation that could impact the safety function(s) of SSCs.

0.9. The term AMP can be applied in a number of ways. It can be applied at the level of an individual SSC to describe the approach to ageing management for that particular SSC; or it can also be used to describe the general or umbrella programme, which covers all of the SSCs at a plant or a number of plants. Within the TPR, the latter is termed the "overall AMP".

0.10. ONR's assessment of ageing management has focussed on assessing the AMPs, covering:

- Overall AMP;
- SSC specific AMPs.

0.11. Chapter 2 of this report describes overall AMPs, focussing mainly on the structure of the AMP and processes used for ageing management. Chapters 3 to 8 look at examples of ageing management to specific SSCs. The SSCs are prescribed by the technical specification and hence are consistent across all of the participating states' national assessment reports.

1 General information

1.1 Nuclear installations identification

1.1. The nuclear installations that fall within the scope of this report are defined in the technical specification as follows:

- Nuclear power plants;
- Research reactors with a power equal to 1 MWth or more;

Research reactors with a power below 1 MWth may also be included on a voluntary basis.

1.2. The national assessment report will cover all nuclear installations that will be:

- Operating on 31 December 2017; or
- Under construction on 31 December 2016.

Those installations which are permanently shutdown and have a regulatory or competent authority obligation to not operate or generate electricity beyond 31 December 2017 are excluded from the TPR process.

1.3. The UK has operated three types of nuclear power plant:

- Magnox
- Advanced Gas-Cooled Reactors (AGR)
- Pressurised Water Reactor (PWR)

The final Magnox Reactor at Wylfa shut down permanently on 31 December 2015 and hence the Magnox reactors are outside the scope of the TPR. This report includes the seven twin reactor AGR stations, the single operating PWR at Sizewell B and the twin PWRs that are under construction at Hinkley Point C.

1.4. Most of the AGR stations were built in pairs and hence the reactor designs of paired stations are almost identical, although there are some minor differences between them. The pairs of stations are:

- Hinkley Point B and Hunterston B
- Hartlepool and Heysham 1
- Heysham 2 and Torness

The design between each set of pairs is different, as is the design of Dungeness B, which was not paired with any other station.

1.5. The UK has only one operating research reactor. This is the Neptune Reactor, a low power test reactor with a power of 100W (see IAEA research reactor database). As such, it is well below the limit for mandatory inclusion in the NAR and is of such a low power that it is not considered appropriate to include it on a voluntary basis.

1.6. The installations included in this report are those in Table 1, which summarises the key parameters for each reactor:

Table 1

Installations included in the UK national assessment report

Station	Type of reactor	Station power output	First operation	Scheduled shutdown
Dungeness B	2 x AGR	1090 MWe	1983	2028
Hinkley Point B	2 x AGR	860 MWe	1976	2023
Hunterston B	2 x AGR	860 MWe	1976	2023
Hartlepool	2 x AGR	1190 MWe	1983	2024
Heysham 1	2 x AGR	1160 MWe	1983	2024
Heysham 2	2 x AGR	1230 MWe	1988	2030
Torness	2 x AGR	1250 MWe	1988	2030
Sizewell B	1 x PWR	1188 MWe	1995	2035
Hinkley Point C	2 x EPR™	3260 MWe	2025	2085

1.2 Process to develop the national assessment report

1.7. There are two licensees for the nine stations that are within the scope of the NAR as follows:

- EDF Energy Nuclear Generation Limited (EDF-NG) is the licensee for the eight operating stations
- EDF Energy New Nuclear Build Generation Company Limited (NNB) is building the new reactors at Hinkley Point C

Whist both of these licensees are part of the EDF Energy Group, they are separate companies and have different management systems.

1.8. The operating reactors were commissioned between 1976 and 1995and the licensee has long established processes for ageing management. Hinkley Point C was granted permission to start construction in March 2017. It is only in the early stages of construction and ageing management has still to be fully defined. It is only at the stage where it is being built into the final detailed design and there are no operational ageing management processes or procedures yet in place. The processes that are used to manage ageing are therefore fundamentally different in the two companies and at different levels of maturity.

1.9. To reflect these differences, each licensee has produced a report that addresses the sections of the NAR that are relevant to them. Whist they have produced separate reports, the two licensees have liaised to ensure that they are consistent in terms of presentation when possible. ONR has provided informal comment on the drafts to ensure that they are consistent with the requirements of the TPR Technical Specification.

1.10. The licensees' reports have been incorporated into the NAR in accordance with the specification. Generally the licensees' contributions have been copied into the NAR without being modified. In some limited cases however, the licensees' contributions to the NAR have been edited to remove excessive detail, improve clarity or so that this report applies a broadly consistent style.

1.11. ONR has assessed ageing management of the reactors using the licensees' reports and its experience gained from its inspection and assessments. The objective of the ONR assessment has been to determine whether ageing is being managed adequately by the licensees. In accordance with the specification, the ONR assessment and conclusions form the final sections of Chapters 2 to 8. ONR's overall assessment and general conclusions on ageing management, including the areas for improvement, are presented in Chapter 9.

1.12. Typical standards that have been used to assess the licensees' ageing management include:

- ONR's Safety Assessment Principles (SAPs, Ref 3)
- ONR's Technical Assessment Guides (TAGs, Ref 4)
- ONR Technical Inspection Guides and (TIGs, Ref 5)
- IAEA Safety Standards (Ref 6):
- Documentation from the IAEA's Programme on International Generic Ageing Lessons Learned (IGALL) (Ref 7)
- Documentation by the Electrical Power Research Institute (EPRI, Ref 8)
- Other appropriate technical standards

1.13. ONR has used its knowledge and experience from a range of its regulatory activities to assess the adequacy of the licensees' ageing management programmes. The specific activities that it has used vary with the aspects of ageing management or the SSC being assessed and include:

- Licence compliance inspections
- Periodic Safety Reviews
- Themed Compliance Inspections
- System Based Inspections
- Outages
- Projects
- Ageing management inspections
- Technical assessments
- Generic Design Assessment (GDA) and licensing assessments
- Specific inspections for the TPR

2 Overall ageing management programme requirements and implementation

2.1 National regulatory framework

2.1.1 Principal legislation for nuclear installations

2.1. The UK's principal legislation for ensuring the safety of nuclear installations consists of the following Acts of Parliament, known as primary legislation:

- Health and Safety at Work etc Act 1974 (Ref 9)
- Energy Act 2013 (Ref 10)
- Nuclear Installations Act 1965 (Ref 11)

Under the UK system of legislation, all of these Acts of Parliament have equal status and all must be complied with. The key features of each of them are summarised in the following three paragraphs.

2.2. Under the Health and Safety at Work etc. Act 1974 (HSWA74), a general duty is placed on all employers and the self-employed to conduct their undertaking in such a way as to ensure, so far as is reasonably practicable (SFAIRP), the health and safety at work of their employees and also those affected by their work activities. This covers both nuclear and conventional health and safety at the licensed sites. SFAIRP is a UK legal term and is equivalent to the guidance term As Low As Reasonably Practicable (ALARP). The term ALARP is used throughout this report.

2.3. The Energy Act 2013 (TEA13) is the legislation that sets the framework for nuclear regulation in the UK. It establishes ONR as a public corporation and defines its purposes and functions. It also allows ONR to appoint inspectors and defines their enforcement powers under this legislation.

2.4. Under the Nuclear Installations Act 1965, as amended (NIA65), no site can be used for the purpose of installing or operating a nuclear installation unless a nuclear site licence is currently in force, granted by ONR. Only a corporate body, such as a registered company or a public body can hold a licence and the licence is not transferable. Under TEA13, those parts of the NIA65 relevant to safety (sections 1, 3–6, 22 and 24A) became relevant statutory provisions of TEA13.

2.5. The key requirement for ensuring nuclear safety is the requirement in HSWA74 to ensure that risks are ALARP. This is a high level non-prescriptive goal, which requires licensees to determine how best to achieve it and to justify their chosen approach. This enables licensees to be innovative and flexible in how they achieve high standards of nuclear safety by implementing arrangements that meet their particular circumstances, taking into account relevant good practices such as those listed in paragraph 1.12.

2.1.2 Licence conditions

2.6. An important provision of NIA65, is that it allows ONR to attach such conditions to the licence as it considers necessary in the interests of safety and radioactive waste management. The powers, to grant a licence and to attach conditions, are delegated by the ONR Board to the ONR Chief Nuclear Inspector.

2.7. ONR has developed 36 standard Licence Conditions (LC - Ref 12) that together form a sound basis for good nuclear safety and radioactive waste management. These address, for example, issues such as operating rules and instructions, maintenance, safety justifications, Periodic Safety Reviews (PSRs), reporting and following-up on events, training and qualification of staff, modification to plant and procedures, independent nuclear safety committees, emergency arrangements, organisational structures and management systems. Several require the licensee to have adequate arrangements to manage changes that may have safety implications.

2.8. The same licence conditions (albeit with minor variations to comply with UK property and ownership laws and variations between English and Scottish environmental legislation) are attached to each site licence issued by ONR.

2.9. The licence conditions mainly set goals and do not prescribe how these goals are to be met. Therefore, each licensee can develop licence condition compliance arrangements that best suit its activities, whilst demonstrating that safety is being managed properly. Under the licence, the licensees have a legal duty to demonstrate adequacy of these arrangements. The arrangements may change as the facility progresses through its life from initial design to final decommissioning. Licensees' compliance with the conditions and with their own compliance arrangements is mandatory. Whilst the system gives flexibility to licensees, it secures high standards in a wide spectrum of nuclear facilities without being prescriptive or requiring detailed rule-making by the regulatory body or parliament.

2.10. TEA13 and HSWA74 provide ONR with a range of powers to enforce the licence conditions and other legal requirements outlined in section 2.1.1. The enforcement powers range from providing advice to influence the licensee through to prosecution. They ensure that ONR can influence improvements and where it identifies shortfalls can take proportionate regulatory action. Details of ONR's enforcement policy are provided on its website (Ref 13).

2.1.3 Regulatory requirements for ageing management

2.11. As noted in paragraph 2.5 the key requirement on the licensee is to ensure that the risk is ALARP. Much of the process of ageing management is addressed through the licence conditions. Ageing management is just one factor that needs to be addressed in the licensee's justification of the safety of their plant. The regulatory requirements are generally not specific to ageing management. The key features of the regulatory requirements with the greatest impact on safe ageing management are summarised below.

2.12. A licence condition requires the licensee to produce safety cases consisting of documentation to justify safety during the design, construction, operation and decommissioning phases of the installation. A safety case provides a written demonstration that relevant standards have been met and that risks have been reduced ALARP. The safety case is not a single document, but the totality of the documentation that justifies safe operation and it evolves during the life of the plant. The safety case must demonstrate the safety of the plant throughout its life including the impact of ageing.

2.13. The licence conditions also require the licensee to identify any limits, conditions and operating instructions that are necessary in the interests of safety. As well as many other aspects, this requires the licensee to implement any arrangements and procedures that are necessary from the safety case to manage ageing and degradation to ensure the safety of the plant.

2.14. A key licence condition for the implementation of ageing management is that requiring examination, inspection, maintenance and testing (EIMT). This requires the licensee to have a maintenance schedule, which identifies all the maintenance (including examination, inspection and testing) identified in the safety case. The licence condition requires the licensee to make and implement arrangements to ensure that all of the necessary maintenance is identified and that it is performed in accordance with the periodicity required by the safety case.

2.15. Other licence conditions contributing to ageing management but which are not described in detail in this report cover the following:

- Incidents on the site;
- Periodic review
- Management systems
- Modification or experiment on existing plant
- Operating instructions
- Periodic shutdown

2.16. The starting point for demonstrating that risks are ALARP and safety is adequate is that the normal requirements of relevant good practice in engineering, operation and safety management are met. This is a fundamental expectation for safety cases. The demonstration should also set out how risk assessments have been used to identify any weaknesses in the proposed facility design and operation, identify where improvements were considered and show that safety is not disproportionately reliant on a small set of particular safety features. The licensee may have to go beyond relevant good practice if the safety case shows that it is reasonably practicable to do so.

2.17. ONR has published Safety Assessment Principles (SAPs – Ref 3), which provide a framework for its inspectors to make consistent regulatory judgements on the safety of licensees' activities. The principles are supported by Technical Assessment Guides (TAGs – Ref 4), and other guidance, to further assist decision making within the nuclear safety regulatory process. The SAPs and TAGs provide ONR's view on what forms relevant good practice and are aligned with relevant national and international standards. Although it is not their prime purpose, they may also provide guidance to designers and duty-holders on the appropriate content of safety cases, clarifying ONR's expectations in this regard.

2.18. The SAPs include a section on ageing and obsolescence within the engineering principles. This includes five principles, although the fifth is related to obsolescence and hence is outside the scope of this report. The four remaining principles relate to ageing and are as follows:

- EAD.1 - The safe working life of structures, systems and components that are important to safety should be evaluated and defined at the design stage.
- EAD.2 - Adequate margins should exist throughout the life of a facility to allow for the effects of materials ageing and degradation processes on structures, systems and components.
- EAD.3 - Where material properties could change with time and affect safety, provision should be made for periodic measurement of the properties.
- EAD.4 - Where parameters relevant to the design of plant could change with time and affect safety, provision should be made for their periodic measurement.

These SAPs deal with ageing across all SSCs. Other parts of the SAPs deal with specific types of SSCs and many of these also include SAPs relating to ageing management of the specific SSCs addressed in this report.

2.19. The TAGs are a suite of documents providing more detailed guidance on specific topics or technical matters. TAG 098 (Ref 14) provides guidance on asset management and focuses on the management of physical plant, including guidance on ageing management. Other TAGs include more specific treatment of ageing where appropriate to their subject.

2.20. Predictions of the effects of ageing and degradation often need to be underpinned by research. Nuclear site licensees are responsible for managing the risks of their operations, and the designers and manufacturers of nuclear plant are responsible under the HSWA74 for undertaking the research necessary to identify and reduce these risks. The Licence Conditions require the licensee to produce safety cases to demonstrate the safety of their operations, so they are responsible for performing any research necessary to substantiate their safety claims.

2.21. ONR also has its own research programme, which is independent of the industry, to inform its own regulatory decision making. Details of the research programme are on the ONR website (Ref 15). The programme incorporates projects related to ageing, including, for example, a project on the ageing of reactor core graphite ageing, which allows ONR to reach independent decisions on this topic, which is described later in this Section 2 of this report.

2.2 International standards

2.2.1 Common elements

2.22. The text in this section of the report has been prepared by the licensees, with only minor changes as described in paragraph 1.9 to 1.10.

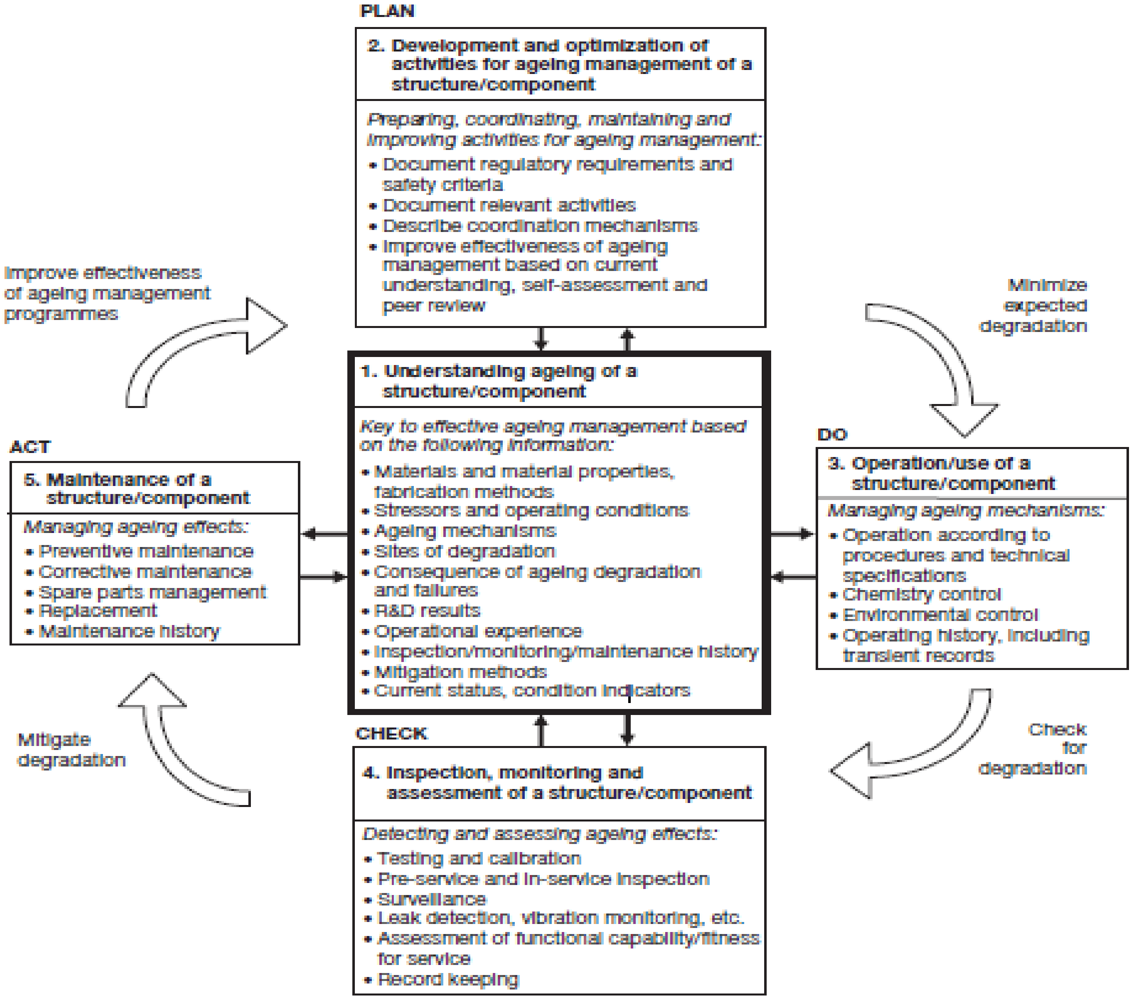

Figure 2.1: IAEA standard for an ageing management programme.

2.23. For both licensees the approach to ageing management is not a stand-alone process; it is integrated into their management systems as described in more detail in Section 2.3.1. The same management systems also enact the licence condition compliance arrangements which fulfil the regulatory requirements for ageing management. The management systems are based on recognised international standards (including relevant IAEA, WENRA and British Standards) and both licensees have fleet-wide, third party certification from an accredited external organisation (Lloyds Register).

2.24. The licensees' approaches to ageing management incorporate guidance provided by IAEA Safety Report Series No. 15 (Ref. 16) and IAEA Safety Guide No. NS-G-2.12 (Ref. 17). They adopt a systematic ageing management process which is an adaptation of Deming's 'Plan-Do-Check-Act' cycle, illustrated by Figure 2.1, and linked to a central requirement to 'Understand' SSC ageing and degradation mechanisms. To reflect this, the process is described in this report as 'Understand-Plan-Do-Check-Act'

2.25. The licensees' approaches to ageing management are also compliant with the ageing management requirements of IAEA Specific Safety Requirements SSR-2/1 (Rev.1) Safety of Nuclear Power Plants: Design (Ref. 18), SSR-2/2 (Rev.1) Safety of Nuclear Power Plants: Commissioning and Operation (Ref. 19), and WENRA SRL I1.1 (Ref. 2).

2.2.2 Operating reactors (EDF-NG)

2.26. The text in this section of the report has been prepared by EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

2.27. In accordance with the footnote to WENRA SRL I1.1, EDF-NG's approach to ageing management provides "*an integrated approach to identifying, analysing, monitoring and taking corrective actions and document the ageing degradation of structures, systems and components*".

2.28. EDF-NG has set down the principles applied when reviewing the adequacy of nuclear safety in the EDF-NG Nuclear Safety Principles (NSPs) for the AGR fleet and Nuclear Safety Assessment Principles (NSAPs) for Sizewell B. The NSPs/NSAPs cover the general basic principles associated with general requirements for the safety case, which includes consideration and management of ageing and degradation. The NSPs/NSAPs are routinely reviewed (every three years) against relevant international standards, including IAEA SSR-2/1 and 2/2 and the relevant WENRA Safety Reference Levels, including WENRA SRL I1.1.

2.29. EDF-NG performs a periodic review of safety for each of its power stations every ten years (paragraph 2.48). This includes a review of ageing, obsolescence and lifetime management against relevant national and international standards, including IAEA Safety Report Series No. 15 and IAEA Safety Guide No. NS-G-2.12.

2.30. Lower level national and international standards that are used to inform relevant company processes, the safety case and company ageing management standards and guidance are discussed later in Section 2.3.2.1 'Ageing assessment', and in the specific examples discussed in Sections 3 to 8. This includes reference to the IAEA International Generic Ageing Lessons Learned (IGALL) programme.

2.2.3 Hinkley Point C (NNB)

2.31. The text in this section of the report has been prepared by the NNB, with only minor changes as described in paragraph 1.9 to 1.10.

2.32. NNB has set down the principles applied when reviewing the adequacy of nuclear safety in its Nuclear Safety Design Assessment Principles (NSDAPs). The NSDAPs cover the general basic principles associated with general requirements for the safety case, which includes consideration and management of ageing and degradation. The NSDAPs were produced taking into account relevant national and international standards, including IAEA SSR-2/1 and the relevant WENRA Safety Reference Levels.

2.3 Description of the overall ageing management programme

2.3.1 Scope of the overall AMP

2.3.1.1 Operating reactors (EDF-NG)

2.33. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

2.34. The following sub-sections provide a description of the overall EDF-NG AMP, which explain how the AMP is enacted by an integrated approach across the EDF-NG management system.

2.35. Two examples of components with significant ageing mechanisms in the AGRs are:

- Graphite core
- Reactor internal components

To illustrate how the AMP is enacted by EDF-NG, these two plant areas are used as examples within this chapter and a running narrative is provided in blue boxes throughout sections 2.3 and 2.4 for both plant areas.

The management system

2.36. The EDF-NG AMP is not a standalone process; it is integrated throughout the EDF-NG Management System. This is illustrated in Figure 2.2 below. It is designed to provide an integrated approach to identifying, analysing, monitoring and taking corrective actions and documenting the ageing degradation of structures, systems and components, in accordance with WENRA SRL I1.1.

2.37. As discussed under paragraph 2.23, the EDF-NG approach to ageing management presented in Figure 2.2 incorporates all elements of the IAEA standard in a cycle that includes 'Understand-Plan-Do-Check-Act', with further description provided in Sections 2.3.2.1, 2.3.3.1 and 2.3.4.1. Additional 'Screening' activities that are used to determine the scope and focus of the programme, which are not presented in Figure 2.2, are described in this section. Under each element Figure 2.2 identifies the most significant company processes that contribute to the element and key contributing components and activities and additional programmes.

2.38. The Figure is not exhaustive, as the activities carried out on all SSCs across all processes would be too numerous and complex to present in one figure, it does however present the most relevant activities. Additionally, all of the activities identified in Figure 2.2 are not applied commonly across all SSCs. An appropriate and proportionate combination of activities are applied depending on factors including the safety significance of the SSC, the ageing mechanisms, the consequence of failure, the type of component (e.g. passive or active), accessibility, etc. This is discussed further in the following sections and the examples provided later in the report.

2.39. The Management System integrates safety, health, environmental, security, quality and economic objectives to ensure that safety is not compromised. It also provides the arrangements that fulfil the relevant Licence Conditions. The Management System is divided into 35 company processes, supported by numerous sub-processes. The key processes that have direct impact on ageing management are summarised below.

2.40. The three EDF-NG processes that have the most significant contribution to ageing management are:

- Maintain Design Integrity (MDI) which manages the production, maintenance and implementation of the safety case. All SSCs are underpinned by a safety case. For significant or life-limiting ageing mechanisms, the safety case may also be supported by additional programme or surveillance arrangements (see paragraph 2.55). These additional arrangements are typically applied to significant irreplaceable passive systems, e.g. the reactor pressure vessel or graphite core.

- Fleet Engineering, Equipment Reliability (ER) which manages the day to day plant condition and performance in accordance with the safety case. The focus of ER is predominantly upon plant and system components where active change in plant configuration and performance can be readily measured and managed.
- Technical Governance (TG) which specifies the engineering standards and guidance to which the plant is designed (influencing the safety case) and controlled (influencing ER). TG supports both the safety case and ER processes.

2.41. It can be seen from Figure 2.2, that these processes are embedded throughout the ageing management cycle. These three areas will form the focus of the remaining commentary in this report, with other processes discussed in paragraph 2.47 below, where relevant to the specific review area.

Overview of company processes and organisational responsibilities

2.42. MDI is EDF-NG's process for the production, maintenance and implementation of the safety case which justifies safety during the design, construction, operation and decommissioning phases of the installation. The safety case demonstrates the safety of the plant throughout its life including the impact of ageing on the plant. The safety case also specifies relevant limits and conditions and other aspects, such as the maintenance and inspection requirements to ensure that ageing and degradation is managed. For significant projects or vulnerable plant areas, the safety case may also include research and development (R&D) in order to underwrite the case. When compiling a safety case it is standard practice to include all reasonably practicable plant inspections where failure of the plant would result in significant consequences. For significant or life-limiting ageing mechanisms, additional programme or surveillance arrangements are put in place (paragraph 2.55). Any challenge to the safety case, either through a revealed plant condition or through a change in understanding (e.g. operating experience, changes to codes and standards, R&D, etc.) would be subject to an immediate assessment using the EDF-NG Safety Case Anomalies Process (SCAP), and appropriate action taken.

2.43. The Station Directors, as the agents for the licensee, are the formal owners of their station safety cases. The Head of Design Authority (an EDF-NG corporate support function) is responsible for maintaining the integrity of the safety case. Where additional ageing management programmes or surveillances are put in place, each has a programme lead or coordinator.

2.44. ER integrates and coordinates a broad range of activities to evaluate station equipment, develop and implement a long-term maintenance plan, monitor equipment performance, and make continuing adjustments to Preventative Maintenance (PM) tasks and frequencies based on equipment operating experience. ER controls the requirements for examination, inspection, maintenance and testing of the plant in accordance with the safety case. The stations' Engineering Managers are responsible for the delivery of ER at their stations.

2.45. TG provides the means by which the company specifies and monitors engineering standards and guidance to which the plant is designed and controlled. TG documents specify design standards, in-service controls, condition monitoring, the testing, maintenance and surveillances that support ER and the safety case. The Chief Engineer in the Design Authority is responsible for the TG documentation that specifies the standards and guidance. The stations' Engineering Managers are responsible for implementation of TG standards on their power stations.

2.46. These three key processes are applied in a proportionate approach, depending on the type of SSC being managed. Table 2 explains the coverage of these processes in relation to the specific SSC examples selected for the TPR.

Table 2: Application of company processes against TPR examples.

<table>
<tr><th>Process</th><th>Electrical Cables (Section 3)</th><th>Concealed Pipework (Section 4)</th><th>Reactor Pressure Vessel (Section 5)</th><th>Primary & Secondary Containment (Section 7)</th><th>Pre-stressed Concrete Pressure Vessel (Section 8)</th></tr>
<tr><td>MDI</td><td colspan="5">All SSCs are underpinned by the safety case, supported by additional programme or surveillance arrangements where required.</td></tr>
<tr><td>ER</td><td colspan="2">ER is applied at System Level across all safety systems containing cabling and concealed pipework.</td><td colspan="3">Limited overview offered by ER process; the main focus of the AMP is through additional programme or surveillance arrangements driven by the safety case.</td></tr>
<tr><td>TG</td><td colspan="5">All examples are subject to TG standards and guidance</td></tr>
</table>

2.47. Other EDF-NG processes that contribute to ageing management include:

- Management of Operations provides the process and standards for the operation of nuclear power stations, including activities which monitor plant indications and conditions and identify degrading plant conditions. This primarily supports the ‘Do’ and ‘Check’ stages of the ageing management cycle (Figure 2.2). The stations’ Operation Managers are responsible for operation activities at their power stations.

- Management of Maintenance is the process that delivers maintenance practices that underpin the safety case, which are controlled by the Maintenance Schedule (MS). This primarily supports the ‘Check’ and ‘Act’ stages of the ageing management cycle (Figure 2.2). The stations’ Maintenance Managers are responsible for maintenance activities at their power stations.

- Outage Management enacts outage planning, preparation, execution and reviews. Many of the plant ageing examination, inspection and testing routines are executed during statutory outages, and are required to satisfy both the safety case requirements and the regulatory requirements to permit return to service. This primarily supports the ‘Plan’ stage of the ageing management cycle (Figure 2.2). The stations’ Outage Managers are responsible for outage activities at their power stations.

- Work Management, working closely with Management of Maintenance and Management of Operations, ER and other processes, provides the process for the scheduling and execution of all PM and non-PM programme activities, surveillance tests and any related support activities. This primarily supports the ‘Plan’ and ‘Act’ stages of the ageing management cycle (Figure 2.2). The managers under the station plant management function are responsible for work management at their power stations.

- The Acquire Goods and Services process ensures that only compliant plant spares are procured, stocked, repaired and installed. This includes arrangements for the control of specification activities associated with the procurement, storage and refurbishment of plant spares, which ensure that the quality meets safety case and licence requirements. This primarily supports the ‘Act’ stage of the ageing management cycle (Figure 2.2). The station Supply Chain Managers are responsible for the day to day supply chain activities at their power stations, and are supported by central Supply Chain managers for corporate-led procurement activities.

- Records Management defines the requirements associated with identifying, collecting, indexing, filing, storing and maintaining records. The types of records are specified for each company process, and are managed in accordance with the Records Management process. Appropriate records are maintained for all stages of the ageing management cycle. Maintenance of adequate records is the responsibility of the line management for each function.

- Asset Management and Business Planning optimise investment in asset safety and performance and managing risk. It includes strategic lifetime planning and short, medium and long term investment planning for the nuclear fleet to enable safe and reliable operation. This primarily supports the 'Understand', 'Plan' and 'Act' stages of the ageing management cycle (Figure 2.2). EDF-NG's corporate Head of Asset Management is responsible for ensuring that a balanced asset investment is maintained.

- Risk Management assesses captures and manages key risk information. It supports the Asset Management process in identifying and prioritising risks associated with plant ageing. This primarily supports the 'Understand', 'Plan' and 'Act' stages of the ageing management cycle (Figure 2.2). The corporate Strategic Risk and Investment Planning Manager is responsible for implementation of Risk Management process.

- Organisational Learning Process (OLP) provides the means to identify, evaluate and address non-conformances, evaluate and make effective use of internal and external operating experience and to evaluate and assess the performance of areas under self-assessment and benchmarking programmes. This includes ageing-related plant events. OLP primarily supports the 'Understand' stage of the ageing management cycle (Figure 2.2), albeit noting that organisational learning is applied to all processes across the whole ageing management cycle. The station Performance Improvement/Continuous Improvement Managers are responsible for the performance of the OLP on their sites, supported by a corporate Fleet Performance Improvement Manager who manages and coordinates the overall programme.

- The Performance Improvement process provides a comprehensive set of Governance and Oversight arrangements which are applied to each company process, known as the 'Fleet Approach'. This approach is used to establish and implement common best international standards and practices for the applicable core processes across the fleet. This supports all processes across the whole ageing management cycle (Figure 2.2). Each company process has a fleet manager or fleet lead, supported by a fleet peer group which includes functional counterparts at each station. The fleet manager and fleet leads are responsible for the development and implementation of their processes.

- The Training & Qualification process incorporates the Systematic Approach to Training. This provides a structured method for producing job performance-based training that addresses personnel and organisational needs and performance deficiencies. The combination of training and the use of Suitably Qualified and Experienced Personnel (SQEP, a Licence Condition requirement) provide the personnel necessary for the development and implementation of the AMP. This underpins all stages of the ageing management cycle (Figure 2.2). Provision of adequately trained and SQEP personnel is the responsibility of the line management for each function, supported by station and corporate training programmes.

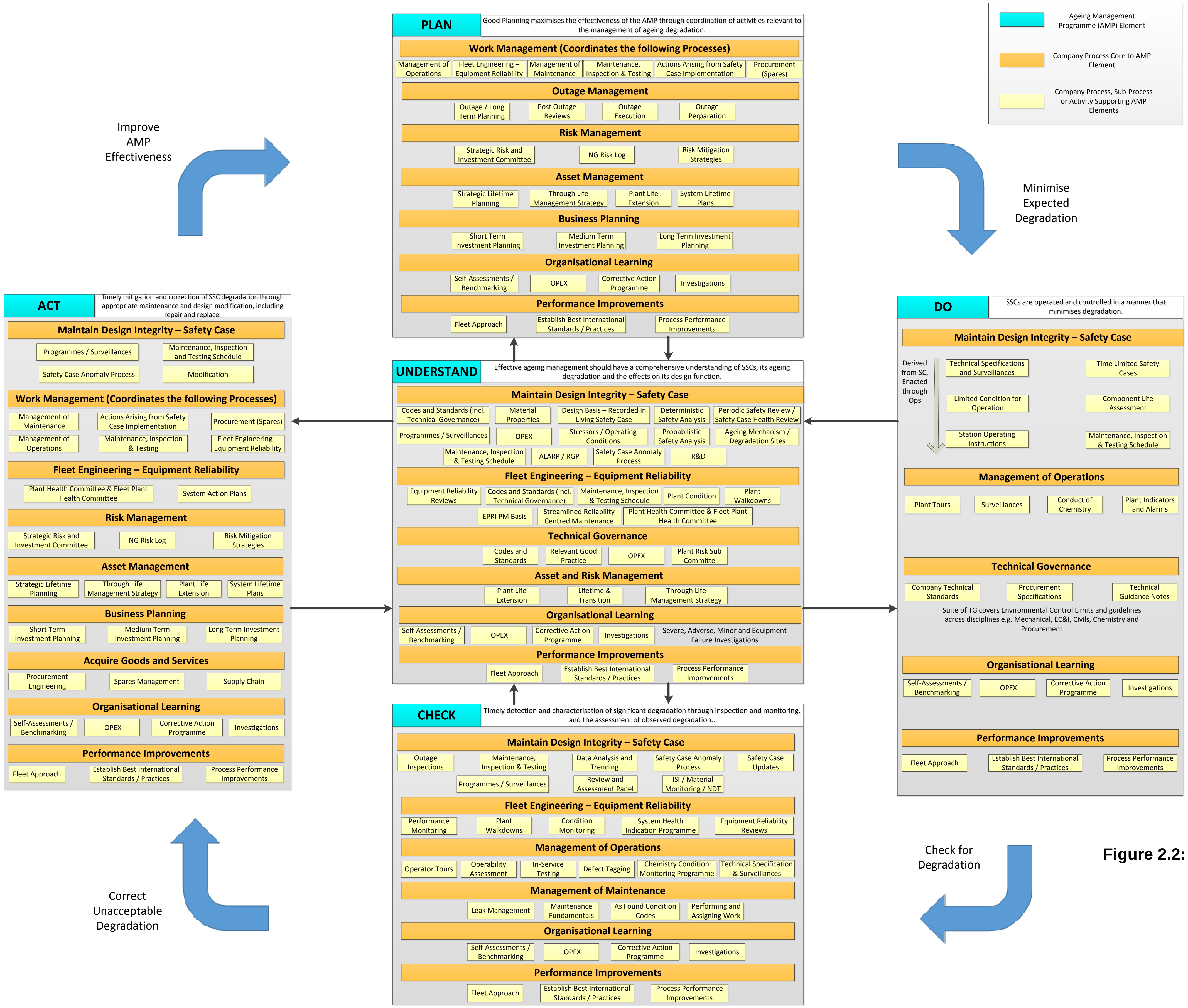

Figure 2.2: EDF NG integrated approach to AM across the Management System.

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2.48. There are also a number of key EDF-NGL sub-processes that interface with these EDF-NG company processes. Examples include:

- Plant Life Extension (PLEX) projects support the Asset Management process. PLEX is supported by the production and implementation of Through Life Management Strategies (TLMS) for key non-replaceable systems (e.g. the boilers, the graphite core, etc.) and other significant plant areas. The TLMS provide holistic plans, covering all people (engineering, supply chain resource etc.) and plant (R&D, operations, inspection, maintenance etc.) issues. They provide a detailed description of threats to future operation, ageing mechanisms, potential work, when it will be required and prospects of operating safely through an extended plant lifetime. This primarily supports the 'Understand', 'Plan' and 'Act' stages of the ageing management cycle (Figure 2.2). The corporate Lifetime Manager within the Lifetime & Transition function is responsible for the PLEX arrangements.

  Replaceable systems are managed under the ER arrangements and System Lifetime Plans (SLPs), under the responsibility of the station Engineering Managers.

- Periodic Safety Reviews (PSR), managed by the MDI process, are performed every ten years for each power station in accordance with the Licence Condition for periodic review. These provide a stand back review of the safety case, the adequacy of the safety-related plant and the effectiveness of the arrangements in place to ensure plant safety. The review includes the extent to which the plant conforms to current national and/or international safety standards and practices. In accordance with the IAEA guidance on PSRs (Ref. 20), the PSR includes a specific review of ageing, obsolescence and lifetime management. This supports the overall review of the AMP (Section 2.4.1). The Head of Design Authority is responsible for delivery of station PSRs.

How the Management System combines to provide an AMP

2.49. Text boxes throughout Section 2.3 supplement Figure 2.2 and describe how the different parts of the Management System provide an integrated approach to ageing management against the IAEA standard. Many of the activities described in the IAEA standard overlap with different parts of the TPR specification. Where relevant, cross reference is made to avoid duplication.

2.50. The EDF Management System approach to the IAEA standard is reported as follows:

Screening	Section 2.3.1 'Scope of the overall AMP'
Understand	Section 2.3.2 'Ageing assessment'
Plan	Section 2.3.3 'Monitoring, testing, sampling and inspection activities'
Do	Section 2.3.3 'Monitoring, testing, sampling and inspection activities'
Check	Section 2.3.3 'Monitoring, testing, sampling and inspection activities'
Act	Section 2.3.4 'Preventative and remedial actions'

Screening (including identification and grouping)

2.51. In line with the IAEA standard (Ref.17) to determine what are effective and practical ageing management measures, it is neither practical nor necessary to evaluate the ageing degradation of all of the SSCs. Establishing the extent of the programme is done through component screening. The following paragraphs identify the key parts of the Management System that enact these activities.

2.52. All SSCs important to nuclear safety are identified in the safety case. The design basis is documented in the safety case and provides the primary means of identification of safety related plant. The extended identification of safety related plant is then subject to overlapping coverage within combined company arrangements to provide compliance across multiple Licence Conditions. Any modification to the plant resulting in addition, change or removal of an SSC would result in an update to the safety case.

2.53. Safety case documentation is made accessible via documents such as the Living Safety Case (for the five older AGRs) and their equivalent, the Safety Case Update Documents and Reference Safety Statement under the Station Safety Report for Heysham 2 and Torness respectively and the Safety Case Users Guide (SCUG) for Sizewell B. These are living documents designed to enhance the accessibility and clarity of the safety case.

2.54. The safety case establishes a MS or Maintenance, Inspection and Tests Schedule (MITS), which mandates periodic inspection of critical components with a safety function and active degradation mechanism.

2.55. For significant or life limiting ageing mechanisms additional programme or surveillance arrangements are put in place. Examples of significant or fleet-wide ageing management related programmes include:

- the AGR graphite cores;
- the AGR Boiler Lifetime Inspection and Monitoring Programme;
- the AGR Component Life Assessment programme for irreplaceable metallic structures;
- the AGR High Temperature Behaviours of Austenitic Stainless Steels (HTBASS) programme;
- the Sizewell B surveillance programmes which include the reactor pressure vessel and containment structures;
- the fleet-wide civil structure inspection programme for compliance with the licence condition for EIMT;
- the fleet-wide corrosion management programme;
- the fleet-wide flow assisted corrosion programme; and,
- the fleet-wide TLMS for key non-replaceable systems.

2.56. The ER programme applies to all power station plant equipment and covers all MS-related systems and components and other non-safety related systems that may adversely affect generation. These are identified at system level using a cross-fleet list of common Electrical Power Research Institute (EPRI) system definitions to inform the stations list of System Health Indicator Programme (SHIP) monitored systems.

2.57. The ER programme then determines the appropriate maintenance strategy based on the significance of the plant item. The approach is based on Streamlined Reliability Centred Maintenance (SRCM). The evaluation includes a step-by-step consideration of the functions that are integral to the operation of the system, the mechanisms of failure of each of these functions, including ageing mechanisms where appropriate, the effects of failure and finally the selection of applicable and effective maintenance tasks to address these identified failures. As part of this evaluation components are classified as critical or non-critical, depending on the consequences of a component failure.

2.58. Other supporting company processes perform grouping of SSCs and documents in accordance with the requirements of the process. For example the TG document suite is based around disciplines, groups of systems, structures or components, which provide an effective means of capturing and disseminating discipline based national and international standards, and imparting knowledge from discipline based Chief Engineers.

2.59. Performance monitoring is carried out at all levels in order to identify gaps and drive forward improvement. Each company process has a range of performance monitoring activities, including internal and external audit, self-assessments, inspections and key performance indicators that are used to inform Governance & Oversight arrangements. Improvements to address performance gaps can result in the identification and formation of additional ageing management programmes, e.g. the fleet-wide corrosion management programme was established to address a deteriorating trend in plant condition, particularly from corrosion on external plant brought about by the coastal environment.

Scope of the Graphite Core AMP

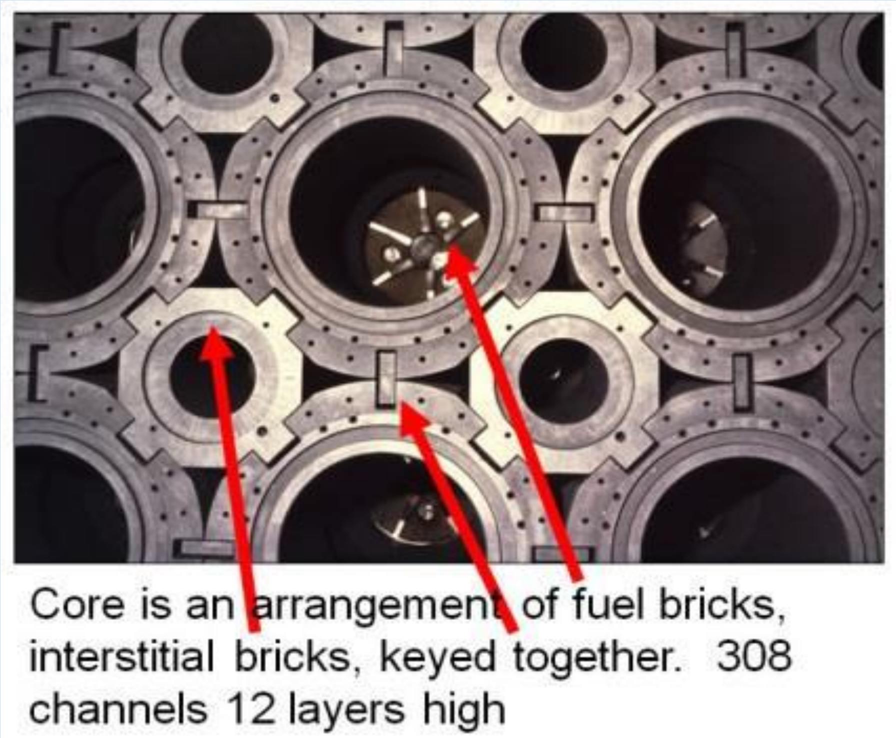
2.60. The AGR graphite cores are of similar, but not identical design, consisting of layers of large graphite bricks forming columns. The columns form 308 to 408 fuel channels (depending on the station). Between these large bricks are smaller square section bricks which form vertical interstitial channels. A number of these are used for control rods which control the overall power of the core. Outside this central region of the core there is further graphite, which acts as a reflector. The fuel and interstitial bricks are joined by a series of graphite ‘keys’ which maintain the overall structure in its nominal position.

2.61. The graphite cores are designed to fulfil a range of functions essential to the safe operation of the reactor. The key requirements are to:

- allow the unimpeded movement of control rods and fuel, including the operation of the secondary and tertiary shutdown systems;
- direct gas flow to ensure adequate cooling of the fuel and core; and,
- provide neutron moderation and thermal inertia.

The first two of these requirements require the core geometry to be maintained, specifically, the fuel and control rod channels to be adequately straight and free from significant distortion and obstructions. The third requires graphite material properties to be maintained at understood levels.

Figure 2.3: Example of graphite core (Hinkley Point B)



2.62. There are two principal ageing and degradation mechanisms affecting the AGR cores, namely: i) graphite weight loss and ii) graphite shrinkage and material property changes. The combined effects are predicted eventually to lead to brick cracking. As these factors may challenge the ability of the core to fulfil its key safety functions there is a limit to the allowable degradation of graphite weight loss and brick cracking. The significance of these issues, and that the graphite cores are irreplaceable (so these limits are likely to determine the actual station operating lives) means that the graphite cores have been identified as an area for a comprehensive ageing management programme.

2.63. The graphite ageing management programme is supported by dedicated groups, which include subject matter experts in graphite properties, core inspections and safety cases. The programme is also supported by a Graphite TLMS to ensure the correct levels of investment are made in the appropriate areas of plant, safety case and personnel to allow the period of the safe and reliable operation of the graphite core to be maximised at each Station.

Scope of the Reactor Internal Components AMP

2.64. The AGR reactor internals describes, essentially with some exceptions, the fixed metallic structures within the pre-stressed concrete pressure vessel, surrounding the graphite core. This includes components such as the core support, the core restraint including the boiler shield wall, the gas baffle, internal pipework, guide tubes, etc. (see Figures 2.4 & 2.5).

2.65. The ongoing structural integrity of these components is important to nuclear safety. The purpose of the reactor internals is to provide the structural geometry (a) to support effective heat exchange from the nuclear fuel through the CO2 primary coolant to the boilers, and (b) to assure safe reactor shutdown. During normal operation this is achieved by forced circulation using the gas circulators although natural circulation is also acceptable when the reactor is in the pressurised, shutdown condition. The cooler gas exiting the boilers is kept apart from the hot gas leaving from the top of the fuel channels by a gas baffle, which takes the form of a steel cylinder, topped with a torispherical dome. The cylinder section also supports half the weight of the boiler and the dome provides access into the graphite core, through guide tubes, for fuel and control rod assemblies. The gas baffle also encloses the boiler shield wall, which acts as a biological shield and forms an integral part of the core restraint system to maintain core geometry. The core itself is supported from below by a lattice grid structure (the diagrid) and support columns, which allow radial expansion and transfer the loads, through the steel liner, to the reinforced concrete pressure vessel.

2.66. There are a number of ageing and degradation mechanisms, some of which are unique to the operating environment of an AGR. The most significant degradation mechanisms are in the form of creep, creep fatigue and low cycle fatigue, which are monitored as a part of the Component Life Assessment (CLA) programme. Other ageing and degradation mechanisms include high cycle fatigue (vibration), irradiation embrittlement, thermal and strain ageing, reheat cracking, oxidation and metal loss, degradation of thermal insulation, carbon deposition, fretting and tribology.

2.67. As these factors may challenge the ability of the core to fulfil its key safety functions there is a limit to the allowable degradation. The significance of these issues, and that in general the reactor internal components are irreplaceable, means that the reactor internals have been identified as an area for a comprehensive ageing management programme.

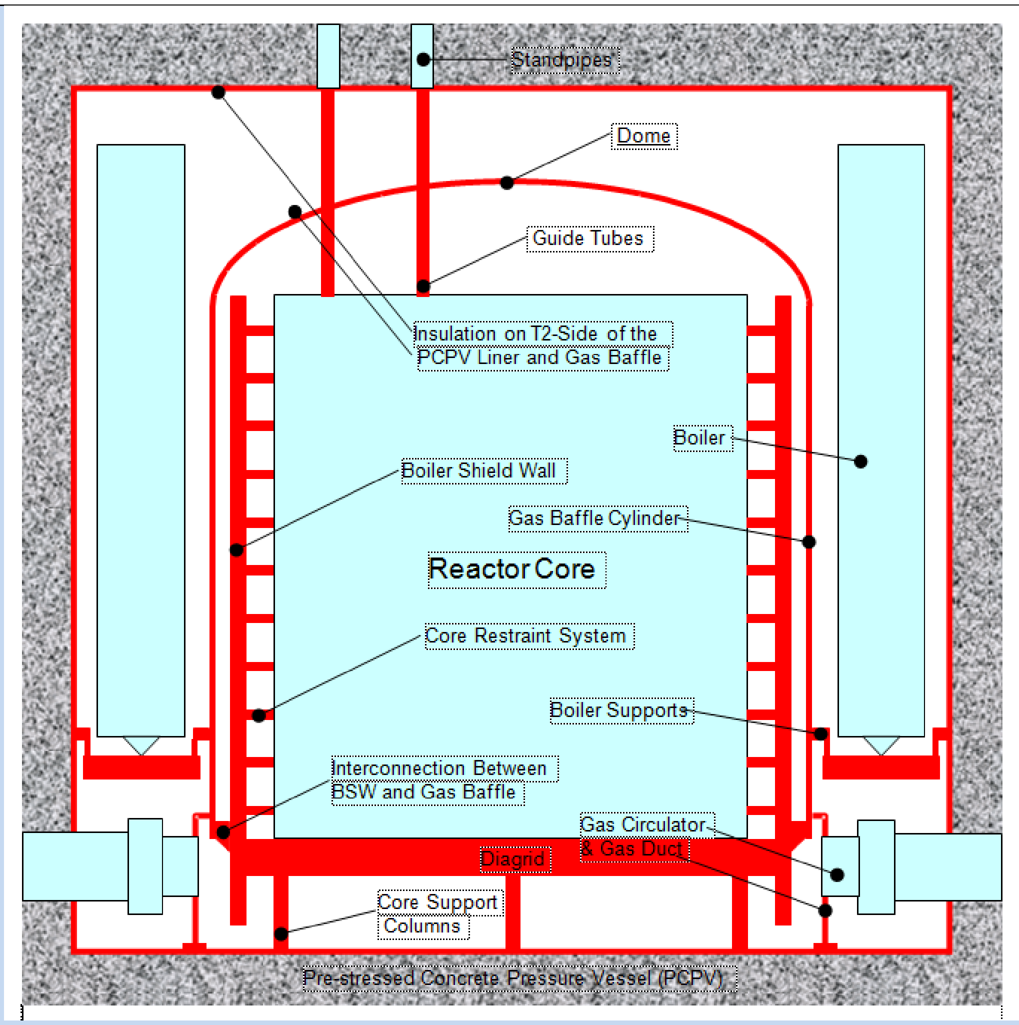
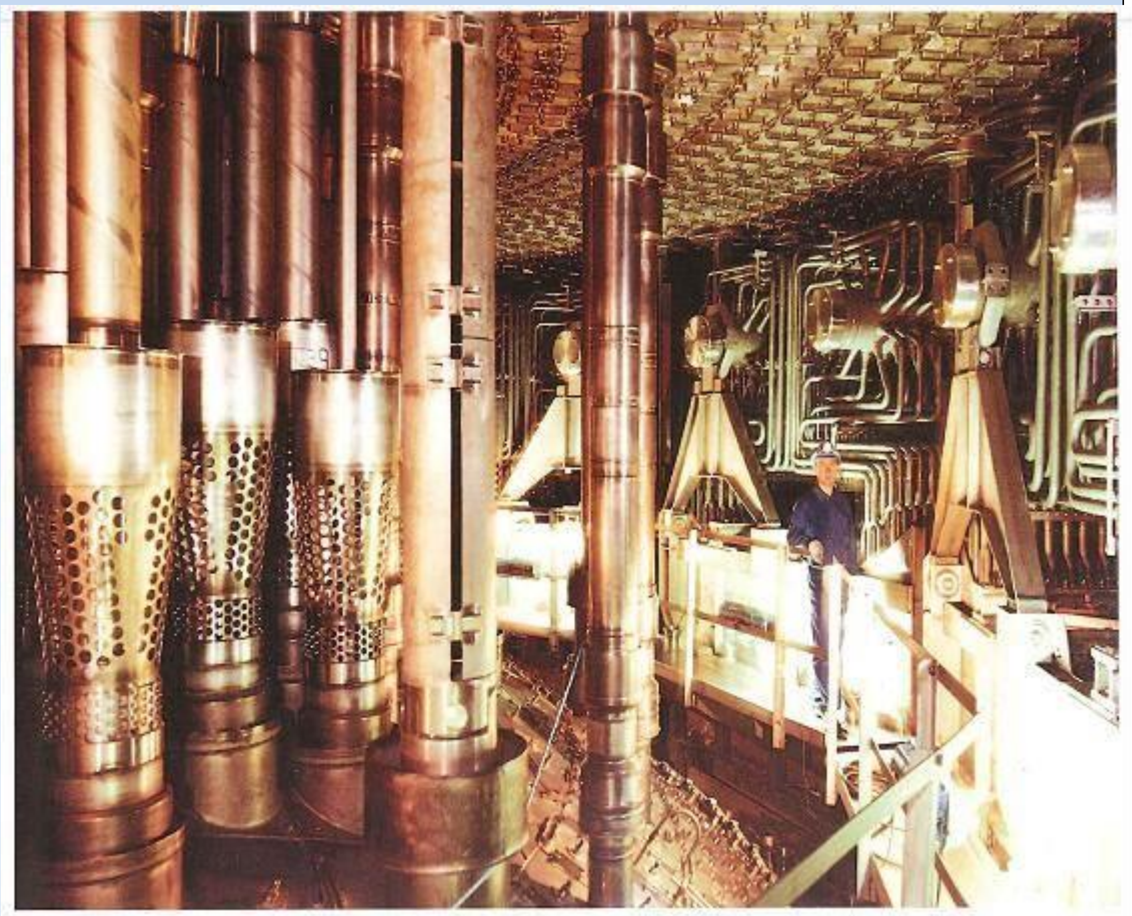


Figure 2.4: Schematic of Hinkley Point B Reactor Internals

2.68. The reactor internal ageing management programme is supported by dedicated groups, which include subject matter experts in material properties, structural and thermal analysis, plant inspections and safety cases. The programme is also supported by dedicated programmes including: the Component Life Assessment (CLA) which determines, monitors and reviews the calculated accumulated creep, fatigue and creep-fatigue damage of AGR boiler and reactor internal components; and, the High Temperature Behaviour of Austenitic Stainless Steels (HTBASS) programme looking at the long-term behaviour of austenitic stainless steels in the reactor environment.

2.69. The programme is also supported by a Reactor Internals TLMS to ensure the correct levels of investment are made in the appropriate areas of plant, safety case and personnel to allow the period of the safe and reliable operation of the reactor core to be maximised at each Station.

Figure 2.5: AGR Reactor internals showing guide tube and internal pipework (picture taken during construction)



2.3.1.2 Hinkley Point C (NNB)

2.70. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

2.71. The NNB AMP is not a standalone process but is implicitly included in the management processes of the Responsible Designer (RD) appointed for the delivery of HPC and NNB's acceptance of the safety case and design deliverables. Due to the current status of the project, the focus is on steps 1 and 2 of the model (Understand – Plan)

2.72. The EDF Group, Direction d'Ingénierie et Projets Nouveaux Nucléaires has been appointed as the Responsible Designer for the EPR™ reactor system and its supporting systems. The relationship is formalised through a contract that includes the scope of responsibility, activities and liabilities in the period up to the commercial operation date and defects period.

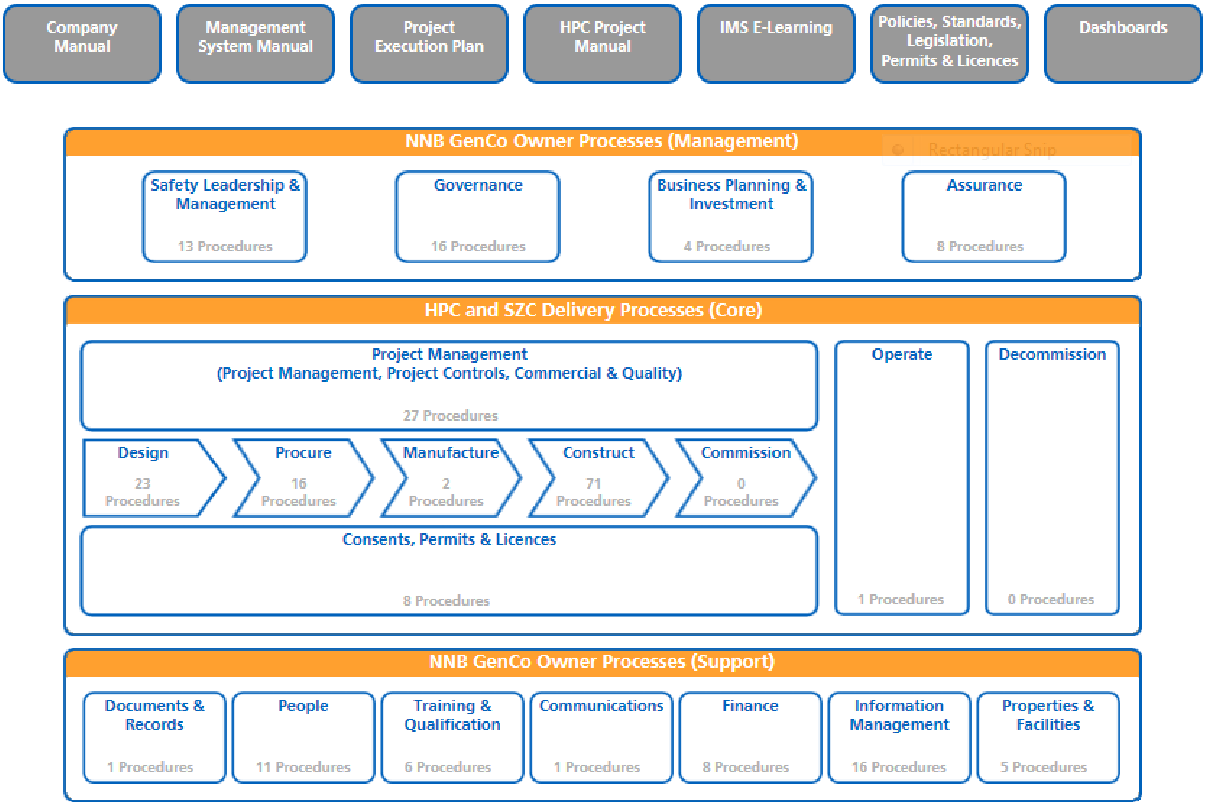


Figure 2.6 – The NNB Management System

2.73. The NNB Management System integrates safety, health, environmental, security, quality and economic objectives to ensure that safety is not compromised. It also provides the arrangements that fulfil the site's Licence Conditions. The current processes within NNB that have the most significant contribution to ageing management are:

- Design – Although NNB does not actually carry out any design of safety significant SSCs it takes responsibility for the acceptance of the safety report (capture design requirements), manages the findings from the Generic Design Assessment (GDA), manages any design change and accepts the design from the Responsible Designer.
- Procure – It approves procurement Technical Specification ensuring the appropriate design requirements have been incorporated.
- Assurance – It Conducts independent assessments, manage and escalate findings as required.

2.3.2 Ageing assessment

2.3.2.1 Operating reactors (EDF-NG)

2.74. The text in this section of the report has been prepared by EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

2.75. EDF-NG's ageing assessment of the TPR is analogous to the 'Understand' stage and is discussed in the following text box.

Understand

2.76. It is a prerequisite for effective ageing management to have a comprehensive understanding of an SSC, its ageing degradation and the effects of this degradation on the SSCs ability to perform its design function. In accordance with the IAEA standard (Ref. 17) this includes an understanding of the design basis (including applicable codes and standards), material properties, the stressors and operating conditions, the ageing mechanisms, the degradation sites, the consequences of degradation, R&D, operating experience, inspection and maintenance history, and the plant condition.

2.77. The safety cases establish many elements of this understanding, starting with the design basis. The safety case is informed by evidence, which can be based on a combination of relevant good practice, standards (including TG documents), material properties, R&D, inspection, analysis (deterministic and probabilistic) and operating experience. This is used to understand the impact of ageing, justify the safe operation of the plant, and manage risks in accordance with ALARP principles. The safety cases justify safety during all phases of the installation, including the design, construction, operation and decommissioning phases. This will define the operating limits and conditions; and, the maintenance testing and inspection programme which may include further analysis of inspection results in order to underwrite continued safe operation. The safety case undergoes periodic review (paragraphs 2.194 and 2.195), which can also lead to developments in the overall understanding. Any challenge to the safety case, either through a revealed plant condition or through a change in understanding, is managed by the SCAP (paragraph 2.146), which may also lead to developments in the overall understanding.

2.78. The process for managing the production of safety cases is consistent with IAEA INSAG 19 (Ref. 21), which is the recognised international guidance on Design Authority requirements.

2.79. The safety cases demonstrate the safety of the plant throughout its life including the impact of ageing on the plant. The safety cases take cognisance of relevant national and international codes and standards for ageing management for the safety discipline or technology under consideration. This is informed by dedicated work supporting the safety case, the ageing management programme or surveillance (paragraph 2.55), or TG documentation covering engineering disciplines or plant (paragraph 2.82).

2.80. A key element to developing understanding is through the application of R&D. Not all ageing issues were known when EDF-NG power stations were designed, built and commissioned. Similar to all other nuclear power plants, emerging ageing degradation issues identified during the operation phase, and improvements in technology, regulations and standards have prompted the need to maintain proactive understanding of material properties and behaviour over time. EDF-NG has an established R&D programme that supports scientific investigations to current and potential ageing management challenges. The R&D programme is achieved through various approaches including: in-house research; alliances with six UK universities; collaboration with the wider EDF group; collaboration with national and international organisations; and, membership in national and international institutions.

2.81. Significant ageing mechanisms are also subject to additional programme or surveillance arrangements in order to fully understand ageing and degradation and to manage the plant and safety case. These arrangements include dedicated teams responsible for managing safety case development, plant inspection, plant modelling, R&D and plant modifications to manage risk in accordance with ALARP.

2.82. The EDF-NG TG process provides the means by which the company specifies and monitors engineering standards for engineering disciplines or groups of common plant or components. Applicable documentation and relevant good practice produced by external industry organisations, other power generators (nuclear and non-nuclear) and suppliers are integrated within the TG document suite. For example, TG documentation on managing Instrumentation and Control ageing is informed by national and international codes, standards and guidance produced by British Standards, IAEA, the International Electrotechnical Commission (IEC) and EPRI. Similarly, TG corrosion management documentation has been compiled using national and international best practice, benchmarks and studies from the IAEA, the Health & Safety Executive, EPRI and the Energy Institute. The complete list of codes and standards is too detailed for this TPR, however, specific examples of national and internal standards are presented in Sections 3 to 8.

2.83. TG documentation covers equivalent standards and guidance to the IAEA's IGALL programme. Whilst the IAEA IGALL programme is not explicitly referenced in the TG or other company process documentation, a comparison has been carried out as part of this review (see Sections 3 to 8).

2.84. TG also takes cognisance of relevant internal and external operating experience, and plant risks which are raised through the EDF-NG Plant Risk Sub Committee.

2.85. Within the ER programme, the SRCM approach (paragraph 2.57) and the selection of efficient and effective maintenance tasks relies upon a thorough understanding of the dominant failure modes and their causes as well as the effectiveness of PM tasks in addressing those causes. The failure causes are determined by considering degradation mechanisms that affect the component (e.g. corrosion, fatigue, etc.). As an aid to this, industry best-practice guidance contained in the EPRI PM Basis Database (Ref. 22) is used. The database provides templates of recommended PM types and frequencies for many component types, taking into account their criticality, operating environment and duty cycle. It also summarises component failure mechanisms and the PM tasks most effective in addressing them. For certain component types both EPRI and EDF France experience are taken into account.

2.86. The ER process was developed in line with international best practice set down in the Institute of Nuclear Power Operations (INPO) AP-913 (Ref. 23). It is also informed at plant level by TG documents and other relevant plant codes and standards.

2.87. The ER programme also includes periodic review of maintenance history and operational trends and data, carried out in Equipment Reliability Reviews (ERRs). Degradation due to ageing that impacts the plant condition or plant performance may be identified by these reviews, and appropriate actions put in place.

2.88. ER obtains an understanding of the actual plant condition through a number of activities. Examples include plant walk downs, operator tours and surveillances, performance parameter trending through SHIP and the outcome from the examination, inspection, maintenance and testing of the plant in accordance with the MS. Oversight of the plant condition and performance is provided by Plant Health Committees and the Fleet Plant Health Committee.

2.89. Operating experience is important in order to develop a comprehensive understanding of ageing degradation. Internal and external operating experience is managed by the Organisation Learning Process (OLP), through the Corrective Action Programme, self-assessments, benchmarking and operating experience. The EDF-

NG operating experience database has captured over 30000 events or topic reports. A search on ‘ageing’ identified over 2000 entries, with more specific searches against discipline or plant ageing typically identifying tens to hundreds of entries. The OLP also provides the company event investigation process. This includes investigations into failures of critical components (known as an Equipment Failure Investigation), which would include failures of plant due to ageing degradation.

2.90. Performance Improvement is carried out at all levels in order to understand performance gaps and the impact of ageing mechanisms. Process level performance monitoring and the ‘fleet approach’, discussed earlier (paragraphs 2.47 and 2.59), is used to inform the overall approach to ageing management. Performance monitoring on structures, systems and components is also carried out, and is discussed later under the ‘Check’ stage.

2.91. The Asset and Risk Management processes provide a number of mechanisms which add to the overall understanding of ageing management, including PLEX projects and the production and implementation of TLMSs (paragraphs 2.47 and 2.48).

Graphite Core & Reactor Internal Component Examples

Graphite Core

2.92. The two principal ageing mechanisms for graphite bricks are:

- Graphite weight loss: This occurs through radiolytic oxidation due to exposure to gamma radiation in a carbon dioxide environment. This reduces the strength and the neutron moderation of the bricks. This condition has to be accounted for in normal operation, reactor faults and hazards, with the most onerous assessed fault condition resulting from water ingress to the core through boiler tube failure.
- Graphite brick cracking: Exposure to fast neutron irradiation will lead to graphite shrinkage and material property changes, which may distort the brick columns. Significantly, the combined effects can cause the bricks to crack, which could challenge the structure of the core. This condition has to be accounted for in normal operation, reactor faults and hazards, with the most onerous condition assessed to result from a severe seismic event.

2.93. EDF-NG’s ageing assessments, which underwrite the safety case for safe operation of the graphite cores, take a multi-legged approach, which includes:-

- understanding the levels of graphite degradation that can be tolerated for brick cracking and graphite weight loss;
- understanding the consequences of degradation coincident with normal operation or reactor faults and hazards;
- predicting with confidence the rate of graphite degradation in brick cracking and weight loss; and,
- performing a suitable inspection programme to determine the condition of the cores with the right level of confidence.

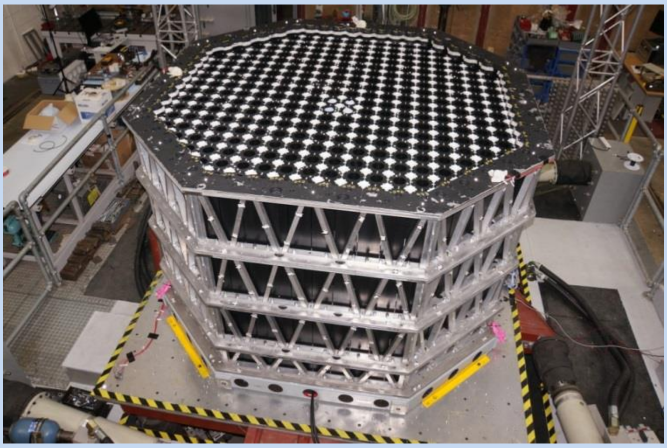
2.94. The safety case is used to justify safety limits on weight loss and brick cracking for each reactor based on extensive research, regular surveillance and analysis of the graphite behaviour, in order to underwrite the multi-legged approach. The first three legs of the assessment are discussed in more detail below. The inspection leg is discussed later in Section 2.3.3.1.

2.95. Weight loss A key part of EDF-NG's programme is the understanding of graphite properties at high neutron doses and high weight losses. This is achieved through subjecting core samples to an advanced experiment using a test reactor in Petten, Netherlands. The results are used to provide a database of material properties and to support a predictive model.

2.96. Graphite properties and brick cracking Irradiation induced creep can reduce internal stresses within a brick, leading to cracking, but this is difficult to model. As a result there are uncertainties associated with the prediction and timing of brick cracking. In order to reduce these uncertainties, further work has been done to understand graphite properties at high doses and high weight losses, including an irradiation creep experiment on graphite core samples. The EDF-NG model and data are also a key part of an IAEA international collaboration on research into graphite creep.

2.97. Tolerance to cracking A significant part of EDF-NG's programme is the understanding of the tolerance to different levels of brick cracking using a combination of advanced computer modelling and experimental support. Brick cracking can occur in normal operation, but the core will be subjected to even greater forces in a severe seismic event, so both need to be modelled. To support this analysis, an experiment on a shaker table of ¼ scale components has been carried out at Bristol University. This complements other experiments that have been carried out on full size single channels.

Figure 2.7: Shaker table of ¼ size graphite core.



2.98. In summary, the AGR graphite core is subject to an extensive inspection, research and modelling programme in order to understand the graphite behaviour, which benefits from the expertise of an EDF-NG team of graphite specialists, along with academics at several universities across the UK and international collaboration.

Ageing Assessment of Reactor Internal Components

2.99. As noted earlier, there are a number of reactor internal component ageing and degradation mechanisms including creep, creep fatigue, low cycle creep fatigue, high cycle fatigue (vibration), irradiation embrittlement, thermal and strain ageing, reheat cracking, oxidation and metal loss, degradation of thermal insulation, carbon deposition, fretting and tribology.

2.100. The ageing assessment which underwrites the safety case for safe operation of the reactor takes a multi-legged approach, which includes:-

- understanding the levels of component degradation that can be tolerated;
- understanding the consequences of degradation coincident with normal operation or reactor faults and hazards;
- predicting with confidence the rate of degradation in components; and,
- performing a suitable inspection programme to determine the condition of reactor components with the right level of confidence.

2.101. The safety case justifies safety limits ageing and degradation for each reactor, based on extensive research and regular surveillance and analysis of the component behaviour in order to underwrite the multi-legged approach. Specific aspects supporting the management of relevant ageing and degradation mechanisms are discussed in more detail below.

2.102. Creep, creep-fatigue & low cycle fatigue is principally but not exclusively driven by changes in temperature (thermal expansion) and pressure, and/or operation at elevated temperature. There are number of components predicted to have creep and fatigue damage before the end of life; the extent to which is subject to detailed analysis under the CLA programme. The CLA programme is designed to provide early warning of the potential development of defects in the internal components of the reactor and boilers due to creep fatigue (including reheat cracking, discussed later). The programme uses the plant operating history to periodically assess past cyclic loading together with a set of anticipated plant cycles and frequencies for predicting end-of-life accumulated damage. The CLA programme is underwritten by an understanding of the materials data and damage prediction, which in turn are underwritten by a comprehensive programme of materials testing, monitoring of reactor operating conditions and plant inspection.

2.103. Concerns were identified regarding the methods used to predict long term creep properties of austenitic stainless steels. The unique combination of CO2 and high operating temperatures results in austenitic steels displaying creep/fatigue damage that may be exacerbated by modification of the surface due to the interaction with environment (particularly the coolant gas). The High Temperature Behaviours of Austenitic Stainless Steels (HTBASS) programme was initiated to address these concerns. The HTBASS programme involves material samples, R&D, plant inspection, plant modelling and safety case analysis in order to set operating and plant life limits. The current understanding is that the reactor internal CLA components are low risk for the current station lifetimes.

2.104. High Cycle Fatigue, i.e. vibration resulting from acoustic excitation and turbulent flow. This issue was carefully considered during the design phase and subject to tests and measurements during commissioning, in order to confirm that the assumptions made during the design phase were conservative. This has been confirmed by subsequent review and inspection during statutory outages.

2.105. Oxidation / Corrosion. Steels in a CO2 environment are subject to corrosion (metal loss), which results in a reduction in the amount of metal in the structure (wall thinning) and also an increase in the overall thickness due to lower density and increased molecular weight of the oxide. This is a well understood process and was a fundamental part of the design process. There is an ongoing programme of monitoring, using a range of specimens and components subjected to reactor environmental conditions. The CLA programme also takes account of metal loss.

2.106. Thermal and Strain Ageing is a degradation mechanism that results in material microstructure changes from prolonged exposure to elevated temperatures. The understanding of thermal and strain ageing is underwritten by an understanding of the materials data which is supported by a comprehensive programme of materials testing and plant inspections.

2.107. Reheat Cracking occurs in welds where the weld is unable to accommodate the strains associated with the creep relaxation of residual stresses at high temperature. Reheat cracking is more prominent in early life as the residual manufacturing stresses are relaxed, but later in life the issue is not the initiation of new cracks but the existence of undiscovered cracks which have been slowly growing. Welds which are inspected in later life and demonstrated to be defect-free are unlikely to generate reheat cracking. The understanding of reheat cracking is informed by the weld type, material, weld methods including the choice of heat treatment, knowledge of operating conditions and plant inspection. Reheat cracking is a low risk for reactor internal components.

2.108. Irradiation causes hardening and embrittlement of carbon manganese steels (notably the diagrid / core support), reducing their resistance to impact loading. Materials other than carbon steels are more resistant to neutron irradiation induced hardening and embrittlement. While this does not affect normal operation, it is more challenging to the safety cases for handling items while the reactor is cool, particularly when considering a potential fault or hazard resulting in a dropped component. Shielding is incorporated into the design of the AGR core where possible to limit the dose to vulnerable components. The ageing assessment is underwritten by an understanding of the doses and the related damage, plant inspection and examination of reactor specimens.

2.109. Other mechanisms such as erosion, fretting, wear, crack growth, insulation permeability, relaxation of guide tube lateral loads, reactor settlement / tilt, carbon deposition and creep relaxation are considered to be lower risk for reactor internal components, but have appropriate work and monitoring to ensure full consideration of operation to planned lives.

2.3.2.2 Hinkley Point C (NNB)

2.110. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

2.111. The Responsible Designer provides design service capability and experience including unique and extensive EPR^{TM} specific knowledge and operational experience gained from the French nuclear fleet and other EPR^{TM} projects to the Hinkley Point C project. It has incorporated this learning and know-how in the plant design. This full capability is being brought to bear and forms the foundations of Hinkley Point C Project. With a design life of 60 years, ageing management is a key consideration in the design of the EPR^{TM}.

Figure 2.8 – HPC's Design Process Overview

2.112. The overall design of HPC is controlled by a Design Process operated by the Responsible Designer and is broken down into three "Levels" of design. The levels are identified as Level 1, Level 2 and Level 3 which represent an increasing amount of detail being added to the design as it progresses.

- - Level 1 Design – Concepts, Codes and Methods
- - Level 2 Design – Systems and Building Layout design
- - Level 3 Design – Detailed design for construction or manufacture

2.113. The Level 1 designer role provides the specifications for the System Requirements and controls a multitude of generic equipment specifications, engineering rules, tools, codes and standards etc. These documents have all evolved throughout the years of operating experience attained by EDF from within the French Fleet and internationally, the documents that hold this information are collectively called the EDF Referentials or Doctrines. These are owned and maintained by EDF's Direction d'Ingénierie et Projets Nouveaux Nucléaires and are used as design references for new plant designs and modifications/refurbishment of the existing fleet.

2.114. The EDF Referentials are supported by documents called Book of Technical Specifications (BTS) and Book of Technical References (BTR). The BTS and BTRs are available and cover many SSCs. For example, there is a BTS for electrical cables, piping systems and various for different aspects of civil engineering. These form the basis of the design work for the EPR™. They have all evolved over the years and have been updated taking into account feedback from French operational experience, R&D and international experience. Typically, they will all be reviewed before the start of a new reactor design, FA3 for example, and periodically or as and when required. They have evolved over time and implicitly include ageing management as the French Nuclear Industry seeks to improve reliability and design the new generation of reactors (EPR™) for a 60 year life. The BTS and BTRs contain elements of the understanding ageing of SSCs from the IAEA safety guide NS-G-2.12 (Ref 17) including feedback from R&D results and operational experience.

2.115. The French nuclear industry also benefits from having its own nuclear design codes (for example RCC-M, RCC-E, ETC-C) which also have been updated taking into account operational feedback and international best practice. RCC-M for example, the code for mechanical components will determine the inspection requirements during plant life, taking into account the effects of ageing, wear out, etc. Nuclear codes form part of the Plan Elements from the IAEA safety guide (Ref 17)

2.116. For HPC, this was a historic position and was recorded as the reference Configuration, based on Flamanville 3 (FA3), and used for the Generic Design Assessment (GDA). During GDA, ageing management was considered and has been reported in the ONR Step 4 Reports (Ref 24). Assessments were undertaken and included the civil, electrical and structural integrity areas and although no major deficiencies were identified, a number of specific Assessment Findings (AF) were raised in the structural integrity area. The assessment, however, did recognise that a dedicated and compressive AMP, comprising all ageing affects for SSCs, could not be found and should be created detailing the programme which will follow the ageing of these components during plant life.

2.117. The Level 2 designer receives the specifications from Level 1 and performs the Level 2 design work. This is the technical work required to develop the system design and plant layout details that ultimately lead to detailed requirements specifications for provision of detailed Civil Engineering Design or Equipment Specifications.

2.118. Equipment specifications will include details of the qualification requirements a piece of equipment needs to be qualified against. As part of the Equipment Qualification (EQ) process plant/equipment that cannot meet the design life of 60 years will be replaced as part of the stations maintenance arrangements.

2.119. The Level 2 design work is passed to the Level 3 “designer” who provides the detailed design documentation. (e.g. detailed equipment design, civil engineering design studies and construction drawings.) For equipment supply contracts, this detailed design would include detailed drawings of the equipment, performance characteristics, spares lists, installation, maintenance, operational requirements etc. that accompany the hardware supplied.

2.120. As part of NNB’s Intelligent Customer (IC) role it reviews and accepts the design by enacting its Review and Acceptance (R&A) process which is managed by the engineering function within NNB. When reviewing deliverables NNB would reference the specifications, the referentials, the safety case, output from GDA, international standards and other relevant references before formally accepting the design deliverables. The definition of the term Design Assurance is depicted by the “V” diagram shown above (Figure 2.8). Design Assurance is considered to be the activities that are undertaken by the contributing project entities to ensure there are no significant errors or inadequacies in the design when considered against the technical specifications.

2.121. NNB’s surveillance is based on a sampling approach focusing on the most nuclear safety significant SSCs as a priority. Factors such as obsolescence, ageing management and equipment qualification will be considered at this point. NNB’s role within the design process is to act as an intelligent customer (IC) on the design deliverables produced by the RD. NNB does not currently design any of the nuclear safety-related SSCs for HPC.

2.122. As part of the R&A process a number of key stakeholders will be part of the review including:

- **Owner’s Engineering** – Responsible for the overall coordination and formalisation of owner’s requests and requirements, including plant performance, operability, maintenance and non-nuclear UK regulation. Coordinates the owner’s surveillance of the design activities undertaken by the Responsible Designer and Industry Partners. Reviews and accepts design deliverables; engaging other key stakeholders, such as Design Authority, Pre-Operations, Health and Safety and Manufacturing Inspection.

- **Design Authority** – Owner of the safety case, they ensure requirements related to the nuclear safety case are defined and adequately addressed within the design, confirming specifications for procurement and conducting surveillance at Site to ensure the design's integrity is intact. The Design Authority leads interactions with the regulators in relation to design.
- **Programme Engineering Lead –** The Programme Engineering Lead is the Programme Manager's primary point of contact for engineering coordination of the programme scope. A Programme Engineering Lead will interact with the RD and NNB Engineering Teams, and is hierarchically attached to the Engineering Directorate and will support associated roles and responsibilities in the Engineering Directorate.
- **Pre-Operations –** Represents the ultimate customer, leading the definition of the operating model and the preparations of the operational organisation, which encompasses:
    - ensuring the design, procurement and construction will enable future performances to exceed the business target;
    - preparing for start-up and future operations by defining the future operating model and development strategy under the governance of the Operational Development Committee and to define the deliverables required from the Project.

2.123. As a site licensee, NNB must demonstrate an appropriately detailed understanding of the EPR™ design as part of its IC role. NNB must provide confidence that the HPC EPR™ design is safe and meets safety and environmental regulations, such as those stipulated in the Environmental Permitting (England and Wales) Regulations 2010 and Licence Condition 14 (LC14), setting out the principles of safety documentation.

2.124. In order to support these criteria, NNB has developed the Nuclear Safety Design Assessment Principles (NSDAPs) to assess the safety aspects of the design. These are in-line with international standards and provide a high level statement of safety criteria. NNB can assess the evolving HPC EPR™ design against the developed NSDAPs to assess the level of compliance of the design.

2.125. The development of both the design of the HPC EPR™ and the accompanying safety case is an iterative process, and the safety case has progressed over time as the design has developed. The latest version of the safety case, Pre-Construction Safety Report 3 (PCSR3), has been assessed by NNB against the NSDAPs. This assessment included consideration of the following section of the NSDAPs relevant to ageing management:

> 6.5.9.2 Ageing Management
>
> 1. Ageing management (AM) should be the measure to control the ageing effects.
> 2. Appropriate margins shall be provided in the design for all SSCs with safety function so as to take into account relevant ageing and wear-out mechanisms and potential age related degradation, in order to ensure the capability of the structure, system or component to perform the necessary safety function throughout its design lifetime.

3. Ageing and wear-out effects in all normal operating conditions, testing, maintenance, maintenance outages, and plant states in a Postulated Initiating Event (PIE) and post-PIE shall also be taken into account.
4. Provision shall also be made for monitoring, testing, sampling and inspection, to assess ageing mechanisms predicted at the design stage and to identify unanticipated behaviour or degradation that may occur in service.

2.126. No gaps in the safety case were identified in NNB's assessment against the NSDAPs associated with ageing management. However, the need to formalise ageing management arrangements at NNB was identified. This need was further reinforced by participation of NNB in various UK fleet experience-sharing events concerning corrosion and ageing management. NNB will formalise the ageing management arrangements in a Corrosion and Ageing Management Strategy report due in early 2018. The strategy will provide some guidance for the NNB engineers and cover aspects including feedback from Nuclear Generation, strategy for paints and coatings management, and recommendation for future corrosion defect management. (complete specification currently in development)

2.127. The EPR^{TM} has been designed and manufactured to a number of well-established international standards and nuclear codes, along with additional UK context where appropriate. The incorporated into These international codes and standards have incorporated the benefit of years of learning from operational experience. The EPR^{TM} is an evolution of French N4s and the German KONVOI PWRs and as such, many years of operational experience and feedback from AMPs have been taken into account in the design. EDF has a large R&D capability which also has been used as an input into the evolution of the EPR^{TM}.

2.3.3 Monitoring, testing, sampling and inspection activities

2.3.3.1 Operating reactors (EDF-NG)

2.128. The text in this section of the report has been prepared by EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

2.129. EDF-NG's monitoring, testing, sampling and inspection activities are analogous to the 'Plan, Do and Check' stages of the IAEA standard. These are discussed in the following text boxes.

Plan

2.130. Good planning maximises the effectiveness of the AMP through coordination of activities relevant to the management of ageing degradation. In accordance with the IAEA standard (Ref. 17) it involves coordinating, integrating and modifying programmes and activities that relate to managing the ageing of SSCs and developing new programme if necessary.

2.131. Work Management, working closely with Management of Maintenance, Management of Operations, Fleet Engineering (ER), Acquire Goods and Services (Procurement) and Outage Management, provides the process for the scheduling and execution of all PM and non-PM programme activities, surveillance tests and any related support activities and corrective actions. This includes activities which deliver the requirements of the MS/MITs. Work Management is also used to schedule and execute actions arising from safety case commitments. The Work Management processes implemented at EDF-NG are based on INPO best practice.

2.132. Outage Management enacts outage planning and preparation, outage execution, and post outage reviews. Many of the plant examination, inspection and testing routines are executed during statutory outages, and are required to satisfy both the safety case requirements and the regulatory requirements to permit return to service.

2.133. The Risk Management, Asset Management and Business Planning processes manage physical assets and asset systems required to operate and maintain the nuclear fleet. They take account of the ageing nature of plant, including the planning requirements for ongoing maintenance. Development of optimum risk mitigation strategies is supported by various tools and processes (e.g. TLMS, PLEX & SLPs). It includes strategic lifetime planning and short, medium and long term investment planning for the nuclear fleet to enable safe and reliable operation. Risk Management includes the means for identifying and managing risks, using the company Risk Log and the development of risk management strategies, under the oversight of Strategic Risk and Investment Committees.

2.134. Organisational Learning (paragraph 2.89) and Performance Improvement (paragraph 2.90) are embedded at each stage of the EDF-NG ageing management cycle in order to support the commitment for continuous improvement of the plant, people and processes.

Do

2.135. In accordance with the IAEA standard (Ref. 17) the plant SSCs are operated and controlled in a manner that minimises degradation. Operating procedures and technical specifications are in place to ensure that operational activities are conducted in a way that minimised ageing degradation.

2.136. The safety case specifies operational documentation in accordance with LC23, which requires operation within limits and conditions that are necessary in the interests of safety. This is implemented by Technical Specifications and Surveillances, Limiting Conditions for Operation and Station Operating Instructions. These compel specific plant and system availability and recovery actions to maintain compliance with the safety case. More relevant to ageing degradation, limits and conditions are also set to maintain the operating conditions within the safety case (e.g. boiler and core component temperature limits, irradiation limits, etc.).

2.137. The operational documentation specified by the safety case also includes the MS, the environmental maintenance schedule and technical specification surveillances, which capture the formal and mandatory requirements in order to maintain compliance with the safety case and site licence conditions, and in particular the licence condition entitled ‘Examination, Inspection, Maintenance and Testing’. These specify regular surveillance, maintenance and testing in order to monitor and understand the condition of the plant.

2.138. Time limited ageing analysis, which includes CLA, are identified in the supporting safety cases and managed by the MDI process, which monitors and manages time limited safety cases. The time limited safety cases are related to plant which is predominantly (but not limited to) passive plant components in the nuclear island, which will undergo quantitative assessment of damaging effects such as creep, fatigue, oxidation and irradiation on an appropriate timescale. Operation of the plant is not allowed beyond a safety case time limit, which ensures that the ageing degradation remains within the safety case limit.

2.139. TG documentation specifies environmental control limits and guidelines across disciplines, e.g. chemistry, mechanical plant, electrical plant, etc. These limits and guidelines are designed to minimise plant degradation, and are implemented by station operations, engineering and chemistry functions.

2.140. Operator activities (carried out under the Management of Operations process) include activities such as plant tours, surveillances, monitoring of plant indications, alarms, leak monitoring and chemistry controls. These assist in the operation and control of plant to either minimise degradation or to identify any degrading plant conditions. These activities are carried out in accordance with operational documentation and procedures.

2.141. Organisational Learning (paragraph 2.89) and Performance Improvement (paragraph 2.90) are embedded at each stage of the EDF-NG ageing management cycle in order to support the commitment for continuous improvement of the plant, people and processes.

Check

2.142. The check activity includes measures to provide timely mitigation of degradation. In accordance with the IAEA standard (Ref. 17), this means the timely detection and characterisation of significant degradation through inspection and monitoring, and the assessment of observed degradation to determine the type and timing of any corrective actions.

2.143. For significant ageing management mechanisms, the safety case specifies the examination, inspection maintenance and testing activities which are captured in the MS or surveillance programme. The safety case also specifies the quality assurance and data collection and review arrangements. Data and supporting analysis will be reported in the safety case in order to justify ongoing plant operation.

2.144. More invasive plant inspection occurs during statutory outages, with the inspection requirements driven by the safety case and specified in the MS or surveillance programme. Material sampling and non-destructive inspections are carried out as part of the ISI programmes. The results are analysed, trended and compared to safety case limits, an example of this is the CLA programme used to assess reactor internal and boiler components (see ‘examples’ commentary). The results are reported in a safety case document justifying the return to service.

2.145. Programmes and surveillances may also have additional arrangements in place to support the assessment and sentencing of inspection results. These typically consist of peer groups or panels of experts who are independent of the day to day programme activities. Examples include the Graphite Assessment Panel and Boiler Assessment Panel for AGRs, the Outage Assessment Panel for AGRs and the Outage Structural Integrity Panel for Sizewell B.

2.146. An examination, inspection, maintenance or testing activity may reveal a plant condition that challenges the ageing and degradation assumptions within the safety case. This will be subjected to an immediate assessment using the EDF-NG Safety Case Anomalies Process (SCAP) and an appropriate action taken. In the most significant events this could require a plant shutdown whilst the design basis is restored, or a modification to the safety case to justify continued operation.

2.147. In addition to the safety case inspection and monitoring activities, defence in depth is provided by the day to day plant condition and performance monitoring performed by ER, Management of Maintenance, Management of Operations and other company processes. These include:

- ER activities such as plant walk downs, performance monitoring and trending, condition monitoring, indirect parameter trending (SHIP) and ER reviews.
- Management of Operations activities such as operator tours, defect tagging, in service testing, surveillances and chemistry controls. If the availability of a SSC is called into question it is subject to an Operability Assessment. The Technical Specifications specify the actions associated with plant unavailability and restoration times.
- Management of Maintenance activities including maintenance, leak management, and recording, reviewing and correcting as found conditions. The maintenance fundamentals and standards and expectations that underpin Management of Maintenance are based on identified best practices from the World Association of Nuclear Operators, (WANO GL 2001-3, Ref. 25) and the Institute of Nuclear Power Operators, (INPO 05-004, Ref. 26).

2.148. Additionally, when performing any type of inspection, any unusual or anomalous observation will be assessed even if the component is not part of the original inspection schedule. A recent example of this is the peripheral shielding brick cracking observed at Torness during inspections focussed on the adjacent core restraint components. Although the peripheral bricks have little structural function, and were not part of the original inspection schedule, cracking was observed resulting in a safety case assessment and further inspection.

2.149. Organisational Learning (paragraph 2.89) and Performance Improvement (paragraph 2.90) are embedded at each stage of the EDF-NG ageing management cycle in order to support the commitment for continuous improvement of the plant, people and processes.

Monitoring, Testing, Sampling and Inspection Activities for the Graphite Core

<u>Operation</u>

2.150. The system availability and operational commitments required by the safety case are governed by EDF-NG's Technical Specifications, Surveillance Requirements and MS. This includes limits on control rod movement, core burn-up, gas baffle differential pressure, coolant gas composition and time constraints on empty channels. MS requirements are placed on taking graphite samples, channel bore monitoring and management of thermocouples.

2.151. TG is in place in order to define the operating window for the coolant chemistry of the AGR primary circuits. This is necessary to minimise the degradation of the core, boilers and other internal structures within the pressure vessel. The primary coolant compositions have been set to address the interlinked issues of carbon deposition, graphite oxidation, steel oxidation, tribology, and radioactive discharges.

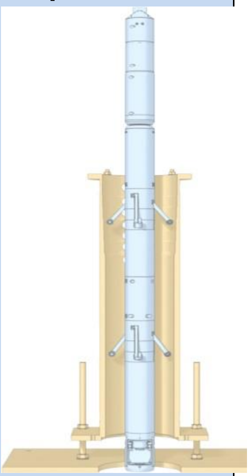
2.152. Routine monitoring of reactor parameters is carried out during normal operation and provides confidence that the graphite core remains capable of meeting its fundamental nuclear safety requirements during the periods between inspection campaigns.

<u>Inspection</u>

2.153. Data on the condition of the graphite core is collected via inspections and sampling in accordance with the stations' MS (in accordance with safety case requirements), and via monitoring of reactor parameters. Core inspections ensure that the condition of the core is established frequently and that sufficient data is gathered during inspection activities to enable adequate future predictions of core and brick behaviour. This comprises video inspections and channel bore measurements of selected channels of the graphite core during periodic shutdowns and interim outages. Following each core inspection campaign, data is used to confirm that the current core state is adequately understood and continues to fall within the constraints of the safety case. Results are also sentenced by a specialist group of peers in a Graphite Assessment Panel. At the end of each periodic shutdown a submission is made in order to both satisfy the safety case requirements and the regulatory requirements to permit return to service. As the cores age, the inspection leg of the safety case becomes more important and frequent.

<u>The NICIE Tool</u>

Figure 2.9: The NICIE inspection tool



2.154. New In-Core Inspection Equipment (NICIE) is the key tool supporting EDF-NG's inspection programme. It is a long cylindrical tool with a video camera, mirror and light positioned at the bottom of the tool. The tool is lowered to the bottom of the fuel channel and is then held in position using two sets of four arms. The tool is pulled up the whole length of the channel to film one section of brick, then it is lowered back down to the bottom of the fuel channel, rotated slightly to the left and then pulled back up to the top of the fuel channel again. This activity is repeated until every part of the fuel brick surface area has been filmed. The individual images are then stitched together to create a full 360^o view of the inside of the fuel channel. If there are any cracks present, they can be clearly seen in this 360^o image.

2.155. The two sets of arms also perform a special measuring function. Inside the NICIE tool are various measuring devices that take measurements of the fuel channel. As the NICIE tool is moved up and down the fuel channel, it records precise dimensions of the channel including any tilts. This information gives a clear indication of any distortion of the fuel channel.

<u>Hybrid Trepanning Tool Unit (HTTU)</u>

2.156. The HTTU is an important piece of inspection equipment, as it allows the removal of samples of the graphite core to be sent off for analysis. Using analysis and data from previous inspections, EDF-NG's Inspection Team plans which channels and bricks in the core will have samples extracted from them. The sample extraction is called 'trepanning'.

2.157. The HTTU is lowered into the fuel channel, stopped and rotated into the precise place where the Inspection Team wish to 'trepan'. A pneumatic jack locks the HTTU into place to prevent the tool from moving and a special drill cuts into the brick, removing a small cylindrical piece of graphite from the fuel brick. A TV camera is then used to check the cut was clean and in the right place.

2.158. The trepanned sample is then sent off to two different laboratories to have their strength and density measured to check the structural integrity of the fuel brick.

Monitoring, Testing, Sampling and Inspection Activities for the Reactor Internal Components

Operation

2.159. The system availability and operational commitments required by the safety case are governed by the Technical Specifications, Surveillance Requirements and MS. This includes limits on various reactor component or location temperatures, pressure, control rod testing, reactor safety circuit settings, coolant gas composition, and refuelling conditions. MS requirements are placed on remote inspections and management of thermocouples.

2.160. TG is in place in order to define the operating window for the coolant chemistry of the AGR primary circuits. This is necessary to minimise the degradation of the core, boilers and other internal structures within the pressure vessel. The primary coolant compositions have been set to address the interlinked issues of carbon deposition, graphite oxidation, steel oxidation, tribology, and radioactive discharges.

2.161. Routine monitoring of reactor parameters is carried out during normal operation and provides confidence that the reactor internals remain capable of meeting their fundamental nuclear safety requirements during the periods between inspection campaigns.

Inspection

2.162. The Central Inspection Group is a team of engineers in the EDF-NG corporate support function focussed on non-destructive testing (NDT) and remote operations activities, which are used during plant inspections carried out to support the safety case.

2.163. In-Service Inspection (ISI) of the reactor internals system is carried out predominantly by remote visual camera inspection at each station. The scope of work and the components to be inspected at each statutory outage are set by the particular requirements of the MSs. In addition to this, the regular CLAs on reactor internal components identify areas that require additional specific inspections. The future scope of ISI is likely to increase due to continued plant ageing and degradation and the increasing focus on plant assessment and knowledge of plant condition to underwrite plant life extension. The need for ISI varies from station to station and is carried out by a range of different equipment.

2.164. All the AGRs were constructed with Oxidation Monitoring Schemes as part of their design. Each Oxidation Monitoring Scheme consisted of coupons, material specimens (larger than coupons) or assemblies (these represent reactor components) placed at various locations to allow exposure of steels to the full range of operational temperatures and reactor primary coolant gas. These in-reactor facilities were generally limited in size, and therefore larger components were exposed in autoclaves attached to the gas by-pass loop. Monitoring equipment was also provided to measure and record the temperatures and gas compositions within the exposure locations. Retrieval and inspection facilities allow the carriers holding samples to be recovered from their exposure location, examined and some returned to their location.

2.165. Inspection of in-circuit coupons and specimens takes place during reactor outages and allows the behaviour of samples to be measured and the trends compared with the original design allowances and the current assessment methodologies. Most samples are replaced following inspection to allow ongoing monitoring however a small number are selected for further, destructive metallography off site. The inspection of the monitoring scheme is a MS item.

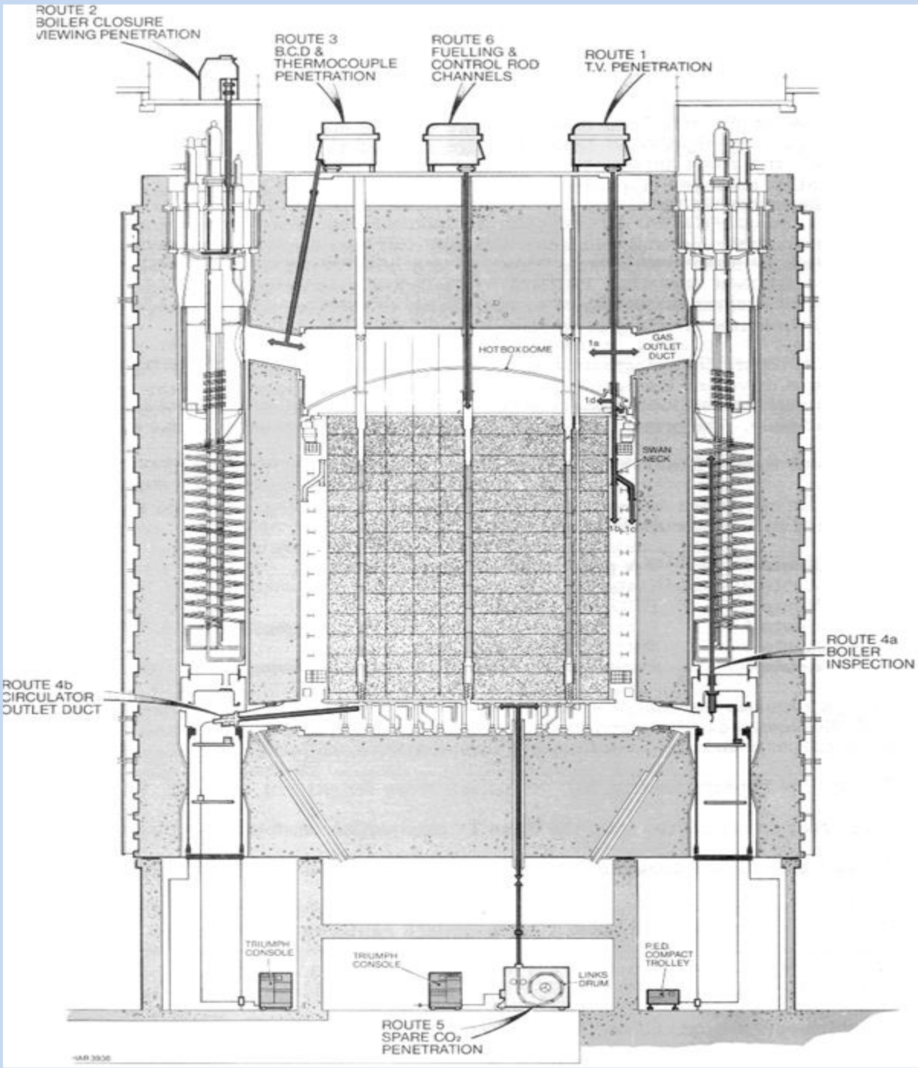


Figure 2.10 Camera Inspection Routes at Hartlepool / Heysham 1

2.3.3.2 Hinkley Point C (NNB)

2.166. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

2.167. Through-life inspection, examination, testing and maintenance will be required for safety critical structures, systems and components (SSC) during the operational life of the UK EPR^{TM} as per Licence Condition arrangements. Full details of the requirements relating to Examination, Inspection, Maintenance, and Testing (EIMT) of the structures will be given in the EIMT programme, which will be finalised at the end of the design process. (It is noted that NNB refers to the Examination, Maintenance, Inspection and Testing or EMIT programme, but this is called the EIMT programme in this report to ensure consistency with the licence condition and the terminology used elsewhere within the report). The appropriate inspection, examination, testing and maintenance arrangements for non-safety-SSCs will also be finalised at the end of the design process.

2.168. Consideration is being given to possible EIMT requirements for HPC during the design stage, the development of which, will take advantage of feedback from the French fleet and Nuclear Generation. Ageing Management will be one of the considerations during the development of the EIMT programme. Inputs into the EIMT programme will be determined from a number of sources including commitments in the safety case (currently PCSR3), and a number of key references which are currently in production or scheduled at a later date. These include Operational Technical Specifications, Periodic Testing documentation, Hazard Technical Specifications and Operational Commitments made to the regulators.

2.169. Another input to the EIMT, where ageing management will be a contributor, will be the development of the Reliability Centred Maintenance (RCM) programme with inputs coming from INPO-AP913 based equipment reliability programmes, OPEX and recommendations from the Original Equipment Manufacturers (OEM).

2.170. The figure below shows this graphically.

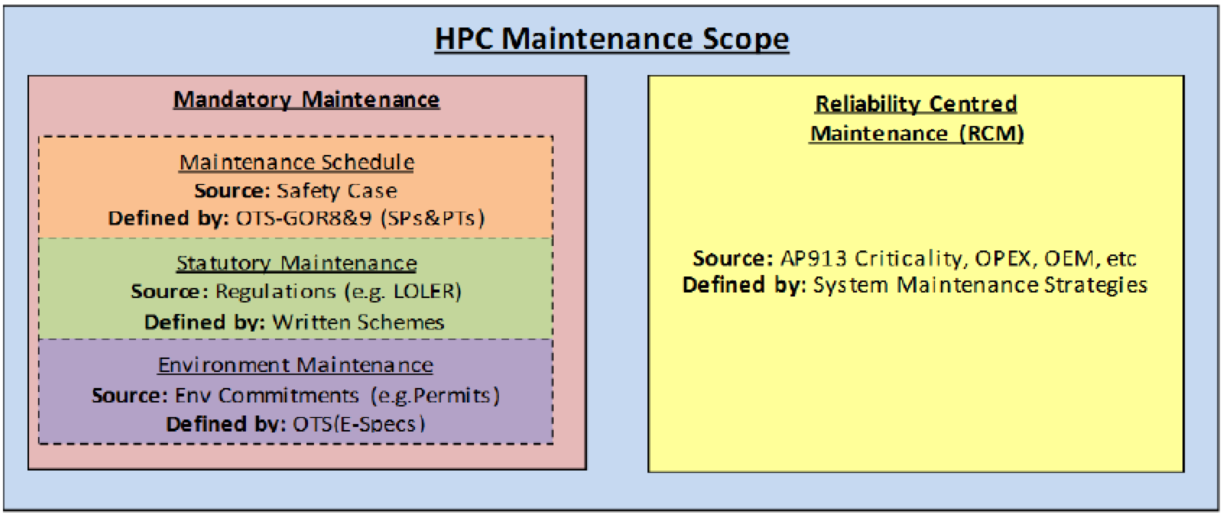


Figure 2.11 HPC Maintenance Scope.

2.3.4 Preventive and remedial actions

2.3.4.1 Operating reactors (EDF-NG)

2.171. The preventative and remedial actions section of the TPR is analogous to the ‘Act’ stage of the IAEA standard, and is discussed in the following text box.

Act

2.172. In accordance with the IAEA standard (Ref. 17) ‘Act’ provides timely mitigation and correction of SSC degradation through appropriate maintenance and design modification, including repair or replacement.

2.173. The safety case establishes a MS or Maintenance, Inspection and Tests Schedule (MITS), which mandates periodic inspection of critical components with a safety function and active degradation mechanism. This ensures timely inspection, maintenance, repair, replacement or modification.

2.174. The design basis may change with time, for example with emergent issues identified during the operational phase, changes in knowledge, or improvements in technology, regulations and standards. These may require a change to the safety case, a plant modification and/or a change to the procedures that are necessary from the safety case to ensure that ageing and degradation is managed to ensure the safety of the plant. As discussed earlier under the ‘Check’ stage, any significant challenges to the safety case sentenced by the Safety Case Anomalies Process (SCAP) may require a plant shutdown whilst the design basis is restored, or a modification to the safety case to justify continued operation. This will be managed in accordance with emergent work processes.

2.175. For significant or life limiting ageing mechanisms additional programme or surveillance arrangements are put in place. These programmes and arrangements provide timely mitigation and correction of ageing and degradation through maintenance or modification.

2.176. The prioritisation and investment for more significant planned work (with respect to risk, scope and cost) is managed by the Risk Management, Asset Management and Business Planning processes (see paragraph 2.133). These processes take account of the ageing nature of plant, including the planning requirements for ongoing maintenance. Development of optimum risk mitigation strategies and strategic lifetime plans are supported by various tools and processes, including:

- ER, e.g. output from the Plant Health Committees, Fleet Plant Health Committee, System Action Plans, etc.;
- The Safety Case, e.g. ALARP plant modifications requiring substantive investment;
- TG, e.g. an understanding of the plant requirements to meet ageing management standards; and,
- PLEX projects and TLMSs.

2.177. Minor corrective work (with respect to scope/cost) and defect correction requiring small investment is managed by the station. The work is prioritised in accordance with its safety significance.

2.178. The scheduling of station work to address preventative and remedial actions is managed by the Work Management process, working closely with Management of Maintenance and Management of Operations and ER. This provides the process for the scheduling and execution of all PM and non-PM programme activities and corrective actions. Work Management is also used to schedule and execute actions arising from safety case commitments (see paragraph 2.131).

2.179. Additional spares are procured to mitigate against unexpected failures, as part of the Acquire Goods and Services process. The Acquire Goods and Services process ensures that only compliant plant spares are procured, stocked, repaired and installed. This includes arrangements for the control of specification activities associated with the supply chain, procurement, storage and refurbishment of plant spares, which ensures that the quality meets safety case and licence requirements.

2.180. Organisational Learning (paragraph 2.89) and Performance Improvement (paragraph 2.90) are embedded at each stage of the EDF-NG ageing management cycle in order to support the commitment for continuous improvement of the plant, people and processes.

Preventive and Remedial Actions for the Graphite Core

2.181. During recent periodic shutdowns at Hinkley Point B there have been a number of observations of keyway root cracks (a crack predicted to be characteristic of later life behaviour). The observations were within the anticipated range for the onset of cracking. Nonetheless, as a result of these observations, the operational period of the affected reactor has been limited, with EDF_NG committing to further core inspections to better underpin predictions of crack progression and rate of crack opening in the core. This understanding will be used to underwrite the longer term safety case.

2.182. The safety case undergoes continuous review and improvement to provide timely mitigation of graphite degradation. Examples of this include significant modifications at the lead stations (Hinkley Point B and Hunterston B) to include the installation of a new set of super-articulated control rods and the commissioning of a new nitrogen injection system (for reactivity hold down). This was introduced to enhance the diversity of the existing shutdown system to cope with the most severe type of seismic event evaluated.

Figure 2.12: Seismically qualified Nitrogen system at Hinkley Point B.

2.183. The super-articulated control rods were implemented by replacing a number of the existing control rods, with rods with enhanced articulation to ensure that they can still be inserted if a channel becomes significantly distorted, and by provision of a new seismically-qualified nitrogen injection system (Nitrogen gas injected into the reactor core from this system absorbs neutrons, stopping the nuclear chain reaction).

Preventive and Remedial Actions for the Reactor Internal Components

2.184. If the damage is predicted by the CLA to reach a components action level by the outage after next, an action must be taken before the next outage. The possible courses of action are (in order of likelihood):

1. Structural Re-Assessment
2. Historical Re-Assessment
3. Consequences of Failure Assessment
4. Inspection
5. Modification of Operating Conditions
6. Repair/Replacement

2.185. Obligations to submit a safety case may arise as a result of these actions. Where reasonable endeavours to reassess a component have been made, but the damage is still predicted to exceed unity within operating life, the component will then fall outside the scope of the CLA process. A specific safety case submission will then be required for the component.

2.186. CLA is regarded as a living process that develops to accommodate changes such as in plant operation, material data, and assessment methodology. Regular CLA reviews are carried out and reported.

2.3.4.2 Hinkley Point C (NNB)

2.187. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

2.188. The preventive actions for controlling and minimising the effects of ageing will be developed later as part of the process of translating the safety case requirements into operational documents. It is expected that this will take advantage of experienced gained in both the French and UK fleets.

2.189. As discussed in section 2.3.3.2, design information is captured from the OEM, OPEX, and the safety case and will be used for the development of the EIMT programme.

2.4 Review and update of the overall AMP

2.4.1 Operating reactors (EDF-NG)

2.190. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

2.191. The review and update of the overall AMP is discussed throughout the 'Understand-Plan-Do-Check-Act' cycle in Section 2.3. The AM arrangements are subject to different types and levels of review by EDF-NG in order to continuously improve both its plant and processes. Examples of specific review arrangements are provided in the following paragraphs, with cross reference to Section 2.3 where appropriate.

Review of the AMP

2.192. The Performance Improvement process provides the tools and arrangements to identify performance gaps and corrective actions for each company process throughout each stage of the 'Understand-Plan-Do-Check-Act' cycle (paragraphs 2.47 & 2.59). An example is the identification and implementation of a corrosion management programme following a deteriorating trend in plant condition.

2.193. As part of the Performance Improvement tools and arrangements, internal quality and internal regulatory audits and surveillances are carried out across all company processes. ER is an example of ongoing company focus area, hence there are additional audit and surveillance programmes supporting this area. External audits and inspections are also carried out, including WANO peer reviews and Lloyds Register accreditation.

2.194. The Periodic Safety Review (PSR) carried out every ten years for each power station includes a specific review of the ageing, obsolescence and lifetime management (paragraph 2.48). PSR has historically identified a number of specific plant ageing issues for enhanced attention.

2.195. In addition to the PSR, EDF-NG reviews the health of each of its safety cases every three years in a Safety Case Health Review. Collectively, these reviews consider the impact of internal and external operating experience, national and international standards, plant performance and events and other relevant information in considering the health of the safety case and potential areas for improvement. A key part of the PSR and Safety Case Health Review is the consideration of ageing and lifetime management.

2.196. Each power station also performs an Annual Review of Safety. Whilst ageing management is not an explicit topic for this review, the performance of programmes managing the impact of ageing (e.g. corrosion management) or plant modifications to mitigate the impact of ageing (e.g. life extension (PLEX) modifications) are included in this annual review. The performance and reliability of the plant are also included.

2.197. Each major ageing management programme (e.g. boilers, graphite, etc.) is subject to a comprehensive set of governance and oversight arrangements, which keep the programmes and underpinning safety cases under continuous review. Management oversight committees and peer groups assess and manage both the technical and project risks and maintain governance and oversight of the lifetime strategies.

Other factors that may require update to the AMP

2.198. In addition to the arrangements to review and improve the ageing management programmes, change may occur due to other reasons. This may be due to a change in the design basis, or a change in understanding resulting from operating experience, revised standards or research. The following paragraphs provide examples of potential changes and how they are managed.

2.199. Any changes to the licensing or regulatory framework which impact the overall ageing management programme would be managed by an appropriate change to the EDF-NG management system. If the change identifies any modification to the design basis, then the safety case would be updated accordingly (see below).

2.200. The design basis may change with time, for example with emergent issues identified during the operational phase, changes in knowledge, or improvements in technology and standards. These may require a change to the safety case, a plant modification and/or a change to the procedures that are necessary from the safety case to ensure that ageing and degradation is managed to ensure the safety of the plant. These are managed in accordance with the MDI modifications process. The modifications process is graded according to the significance and potential consequences of the modification. Significant changes are submitted to an internal nuclear safety committee for advice and consideration and to the ONR for permissioning.

2.201. An examination, inspection, maintenance or testing activity may reveal a plant condition that challenges the ageing and degradation assumptions within the safety case. Additionally challenges may arise from internal or external operating experience, research or a change to a standard. Any challenge to the safety case will be subjected to an assessment using the SCAP (see paragraph 2.146) and appropriate action taken. In the most significant events this could require a plant shutdown whilst the design basis is restored, or a modification to the safety case to justify continued operation.

2.202. The design basis is also defined in a number of areas by time limited safety cases (TLSC) (see paragraph 2.138). Where ageing analysis is time limited due to the extent of knowledge of the mechanism, or the safety arguments are time limited based on an understanding of an ageing mechanism, the time limits are identified in the safety case. These limits are made visible and are managed by the MDI process and associated operational documentation.

2.203. TLSC are related to plant which is predominantly (but not limited to) passive plant components in the nuclear island, which will undergo quantitative assessment of damaging effects such as creep, fatigue, oxidation and irradiation on an appropriate timescale. This is typically presented as either a degradation limit (e.g. core irradiation), an operating limit (e.g. valid for a number of cycles or specified operations hours) or as a defined time period. The ongoing integrity of these cases and their associated plant is assured by the MDI process. The Component Life Assessment (CLA) is a key component of this for AGR reactor and boiler components.

2.204. Internal and external operating experience may provide evidence that is relevant to ageing management. The OLP provides the means to identify evaluate and implement operating experience (see paragraphs 2.47 and 2.89). Any operating experience that challenges the safety case will be subject to the SCAP (paragraphs 2.146 and 2.201). However, operating experience also forms an important basis for the ongoing review and improvement of ageing management programmes and processes. It is used routinely in the production of safety cases or other relevant documents, e.g. TG documentation.

2.205. R&D also provides evidence that is relevant to ageing management (paragraph 2.80). For significant ageing mechanisms or vulnerable plant areas, ageing management programmes identify and include comprehensive ground-breaking R&D in order to underwrite ageing and degradation and other modelling assumptions (see graphite and reactor internal component examples). The safety case and ageing management programmes are regularly reviewed against the output from R&D. Any results that challenge the safety case are subject to the SCAP (paragraphs 2.146 and 2.201).

2.206. All TG documents are subject to a three yearly review, which includes assessment of any relevant internal and external operating experience and changes to national or international standards.

Update of the AMP

2.207. The integrated approach to ageing management across the EDF-NG Management System, and the mechanisms in place to provide continuous review and feedback, result in a continuous cycle of review and update to the ageing management arrangements. Mechanisms are in place to identify and prioritise changes, in accordance with their significance, such that:-

- Significant issues are dealt with as a priority. Events, operating experience, research results, inspection results, etc. can all initiate SCAP where the underlying safety case is challenged (paragraphs 2.146 and 2.201). The resulting action may include an update to the AMP in order to restore the safety case (e.g. there may be a revision to the MS to change the frequency or type of plant inspection or testing).
- There is ongoing routine monitoring and improvement of company processes and the SSCs being managed by the processes (paragraph 2.59). There is also ongoing review of significant ageing management programmes (paragraph 2.55), which can result in an update to the AMP.
- Periodic reviews are carried out across many aspects which support the ageing management arrangements, e.g.:

- company controlled documents are updated every three years, which includes the Nuclear Safety Principles (paragraph 2.28) and TG documentation (paragraph 2.82);
- Safety Case Health Reviews and Equipment Reliability Reviews are typically carried out on a three yearly cycle (paragraphs 2.195 and 2.87);
- the PSR is carried out for each station every ten years (paragraph 2.48); and,
- surveillance programmes are reviewed at defined frequencies.

All of these activities can result in updates to the AMP.

Graphite Core and Reactor Internal Component Examples

Graphite Core

2.208. The graphite safety cases and associated ageing management programmes continue to undergo review and development. Key to the development of the safety cases is the ALARP management of risk, and the implementation of ALARP measures. Work continues to better understand the degradation mechanisms and any associated uncertainties with further development of materials tests, modelling and experiments. Better inspection techniques are also being developed; an example is provided below.

2.209. Longer term safety cases will also continue to consider ALARP modifications to mitigate the impact of graphite degradation, and to develop appropriate safety arguments to claim the benefit of these modifications.

2.210. Safety Case Health Reviews and PSR also continue to provide periodic stand-back reviews of the graphite safety case and ageing management programme, notwithstanding that there is a comprehensive level of governance and oversight applied to the graphite programme at all times.

Eddy Current Inspection Tool

2.211. As a result of the ongoing review of the graphite AMP, there are a number of new inspection techniques being developed by EDF-NG to improve understanding of the evolution of the graphite core. For example, the eddy current tool is in an advanced state of development. This tool measures the electrical conductivity variations within the fuel channel bricks and is used to ascertain the density of the graphite. In the middle of the tool, there are three ‘measuring heads’ which operate on a similar principle to metal detectors. The tool is lowered to the bottom of the fuel channel and then the heads are spun 360^0 degrees whilst being slowly raised to the top of the channel. This helical scan determines the density variations within the whole fuel channel and is used to provide an indication of graphite weight loss.

Figure 2.13: The Eddy Current Tool

Reactor internal components

2.212. Similar to the graphite safety cases, EDF-NG's reactor internal safety cases and associated ageing management programmes continue to undergo review and development. Again, key to the development of the safety cases is the ALARP management of risk, and the implementation of ALARP measures. Work continues to better understand the degradation mechanisms and any associated uncertainties with further development of materials tests, modelling and experiments. Better inspection techniques are also being developed for reactor internal inspections, an example is provided below.

2.213. Reactor internals safety cases in general cover the lifetime of the plant, but in a number of areas time limited safety cases are in place. Updates to these cases are managed by the MDI process. The development of these cases continues to consider ALARP measures to mitigate the impact of ageing and degradation.

2.214. Safety Case Health Reviews and PSR also continues to provide periodic stand back reviews of the reactor internals safety case and ageing management programme, notwithstanding that there is a comprehensive level of governance and oversight applied to the reactor internals programme at all times.

Remote manipulator for inaccessible areas including the reactor vessel.

2.215. As a result of the ongoing review of remote inspection, a robotic snake arm has been developed that can potentially be used as a common manipulator platform to increase the inspection capabilities of the nuclear fleet. With the different reactor designs and access routes, a common, flexible deployment platform could be configured for each inspection and perform a variety of tasks including: visual inspections, ultrasonic inspection, sampling, installation and repairs.

Figure 2.14: Laser snake arm

2.4.2 Hinkley Point C (NNB)

2.216. NNB advises that these aspects will be developed later for EPR™ at Hinkley Point C.

2.5 Licensees' experience of application of the overall AMP

2.5.1 Operating reactors (EDF-NG)

2.217. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

2.218. The EDF-NG Management System, and the supporting arrangements and organisation, undergo continuous review and improvement (paragraphs 2.47 and 2.59). As the EDF-NG AMP is integrated throughout the Management System, it also undergoes continuous review and improvement. A number of the key changes over the last decade and a half are listed below:-

- In 2003, the company launched the Performance Improvement Programme (PIP), a vehicle to address overall performance shortfalls. The Performance Improvement Plan proposed a significant shift in the investment strategy to support continuous improvement with a clear intent to bridge evident performance gaps and align company performance with world best practices.
- A core part of this was the establishment of the 'Fleet Approach' (paragraph 2.47). This is based on a Fleet Management organisation with responsibility for establishing best practice standards across the fleet for core processes and programmes such as Operations, Engineering, Maintenance, Supply Chain, Fuel Route etc. This is led, for each core process, by a central Fleet Manager supported by their functional counterpart from each power station to form a 'peer group'. The peer group develops long term and shorter term improvement plans to close any gaps to world best practice and performance in their area of responsibility.
- In 2009 British Energy was bought by EDF Group. Reinforcement of the Management System, and additional controls, governance and oversight arrangements gave increased focus to seek better safety and operational performance.
- The development of a clear lifetime strategy has been a major improvement in the EDF-NG arrangements. The PLEX approach (see paragraph 2.48) and a clear station lifetime strategy via TLMS have provided clarity and visibility of the key components required to achieve the proposed lifetime.
- The ageing management arrangements across the EDF-NG fleet have undergone significant improvements over this period. Primarily as a result of the changes discussed in the previous bullet points, but also as a result of specific focus areas including ER and TG.

2.219. The fundamentals of the WENRA Safety Reference Levels for ageing management are built within the structure of EDF-NG with oversight provided by senior management.

2.220. Despite the EDF-NG fleet reaching a mature operating age at which ageing and degradation becomes more challenging, the overall fleet safety performance and plant reliability has significantly improved over the last decade, as evidenced by key performance indicators and feedback from WANO peer reviews. However, internal and external oversight, which includes periodic safety review, continue to identify areas for improvement, in line with the commitment for continuous improvement.

2.221. There have been a number of challenges associated with ageing and degradation across the fleet, many of which were anticipated, revealed and managed as part of an AMP. There have also been a small number of unanticipated ageing issues. Examples of both types include:-

- AGR boilers: AGR boiler ageing and degradation is subject to a comprehensive dedicated AMP (paragraph 2.55). The onset of creep/fatigue cracking in Hinkley Point B and Hunterston B boilers (the mechanism is discussed in Section 2.3.2.1) has led to extended outages to carry out additional repairs and inspections, and a reduction in plant load in order to reduce component temperatures.
- AGR graphite cores: The AGR graphite cores are also subject to a comprehensive dedicated AMP (paragraph 2.55 and examples discussed in Section 2.3). Keyway root cracks have been identified in the lead station slightly earlier than expected, but within the anticipated range for the onset of cracking. Nonetheless, as a result of these observations, the operational period of the affected reactor has been limited, with EDF-NG committing to further core inspections to better underpin predictions of crack progression and rate of crack opening in the core. This understanding will be used to underwrite the longer term safety case
- There have been several events over the last decade involving leaks from, and failures of, concealed pipework (see Section 4). A fleet-wide project has been established covering all concealed systems, including cables, civil structures and pipework for systematically identifying and inspecting all concealed nuclear safety related plant in order to establish its condition and to take any remedial actions necessary. A risk-based inspection and repair strategy, based on the potential safety significance of each concealed system, is now in place.
- Internal and external oversight, plant inspection and a number of pipe failure events have identified a shortfall in the management of corrosion. External systems exposed to the corrosive effects of salt in the atmosphere, and in particular where corrosion has occurred under insulation or lagging, have been particularly vulnerable. A fleet-wide corrosion management programme has been established. As with the concealed system programme, a risk-based inspection and repair strategy, based on the potential safety significance of each system, is now in place.

EDF-NG's integrated approach continues to identify, analyse, monitor and take corrective actions in order to manage ageing and degradation.

Recommendations by EDF-NG

2.222. Although IGALL documentation has not been formally incorporated within EDF-NG's Technical Governance (TG) arrangements, the TG has incorporated equivalent standards and guidance. The specific SSCs identified for review in Sections 3 to 8 have considered relevant IGALL documentation, and have not identified any gaps or shortfalls. Nevertheless, EDF-NG recommends that the periodic review and update of EDF-NG's TG documents include review of relevant IGALL documents. EDF-NG plans to update its TG document review management process by 28 February 2018 to include the requirement to review IGALL during the three-yearly cycle of TG document updates.

2.223. Although ageing management is established throughout the EDF-NG Management System, there is no single company document describing this approach. EDF-NG therefore recommends that the description of the EDF-NG ageing management arrangements, described in Section 2 of this report, be issued in a formal company guidance document. EDF-NG advises that this will be captured within its MDI process and will be issued by 31 March 2018.

2.5.2 Hinkley Point C (NNB)

2.224. NNB advises that this is not applicable for EPR™ at Hinkley Point C as both units are still at the construction stage.

2.6 Regulatory oversight process

2.225. ONR provides regulatory oversight of licensees' activities through its inspection and assessment functions. ONR's inspectors, which include its technical specialists, perform a number of functions which are used to oversee ageing management. The key ones are described below.

2.226. For each reactor site there is a nominated site inspector. The site inspector takes the lead on ensuring that the licensee is compliant with its site licence and in particular the conditions attached to it. Inspections can be either planned or reactive. The latter are usually in response to an incident or a finding from an inspection or assessment. There are two main types of planned inspection which are used to confirm that the AMP is still valid:

- Licence Compliance inspections – the site inspector, with support from other inspectors within ONR where appropriate, will inspect the licensee's arrangements for a particular licence condition and ensure that they are being adequately implemented on the plant. ONR has Technical Inspection Guides for its inspectors (TIGs – Ref 5), with each providing guidance on a particular licence condition. Taking an example from section 2.1.2, which is key to ageing management, the site inspector will periodically inspect the arrangements for the licence condition on examination, inspection, maintenance and testing. During these inspections, the site inspector may for example review the licensee's arrangements, witness maintenance activities, discuss application of the arrangements with plant personnel or review maintenance records or instructions.
- System Based Inspections – Each station has been divided into about 30 systems and ONR inspects compliance with the safety case, including ageing management aspects, for about six of these systems a year, so that compliance with the safety case for the whole plant is inspected over a five year cycle. These inspections start from the safety case and consider how it has been implemented on the plant. They look at limits and conditions, operations, and maintenance including inspection of plant. They include reviews of relevant incidents on the plant, records to demonstrate continued compliance with the safety case and inspection of the physical plant. Any issues with ageing management for the system are likely to be picked up during the plant inspection or the review of the licensee's inspection records. The overall output of the inspection is whether the safety case, which includes the AMP, has been adequately implemented on the site, together with individual inspection ratings for the licence conditions looked at.

2.227. In addition to the compliance inspections in the previous paragraph, ONR also undertakes thematic inspections. These look at a specific theme across a number of, and in some cases all, the reactor sites. There can be many reasons for selecting a theme, including the response to ONR strategic themes, incidents or inspection findings and these inspections can cover a broad range of topics including ageing.

2.228. The licence condition on incidents on the site requires the licensee to have adequate arrangement to notify appropriate bodies, record, investigate and report incidents on the site. The most significant incidents are reported to ONR. ONR reviews any reports that the licensee sends as notifications and routinely samples and inspects others to confirm that there are no unexpected problems, for example, due to ageing.

2.229. The site licence conditions require the licensee to have arrangements for controlling modifications, which includes alterations to plant and safety cases. These arrangements require review of any changes and in some cases formal agreement to the modification by ONR. ONR assesses those safety cases where its permission is required on a sampling basis, and may also elect to assess less significant safety cases. As part of these assessments, ONR will consider whether the licensee has taken full account of the impact of the modification on ageing of the plant.

2.230. Each AGR has a statutory outage every three years and the PWR at Sizewell B has a refuelling outage every 18 months. During these outages the licensee inspects the plant in accordance with its Maintenance Schedule and one of the objectives of this is to confirm that ageing is proceeding as expected. Hence this is an important element of ensuring that the AMP remains valid. To this end ONR also inspects the licensee's work during the outage and assesses the results of its inspections where appropriate. Restarts from statutory outages require ONR's permission (termed a Consent). ONR only issues Consents for restarts where we are satisfied that our assessment has concluded that the licensee's safety case, which includes its AMP, is still valid, adequate and properly implemented.

2.231. In line with established international practice, ONR uses the Licence Conditions to require the licensee to undertake a Periodic Reviews of Safety (PSR) for each site every ten years. Here the licensee has to review the impact of ageing on nuclear safety and demonstrate that the plant is safe to operate up to the next PSR. ONR undertakes comprehensive assessments of the licensee's PSRs. It will only confirm that continued operation is acceptable if the licensee has demonstrated that it has an adequate strategy for dealing with ageing and obsolescence up until the next PSR.

2.232. ONR publishes reports of its inspection activities and key regulatory decisions on its web site. For inspections, including those described earlier in this section, it publishes the executive summaries of its inspection reports, which it calls intervention records (Ref 27). For a regulatory decision, including Consent for restart and continued operation following a PSR, it publishes the Project Assessment Report which justifies the decision (Ref 28).

2.7 Regulator's assessment of the overall ageing management programme and conclusions

2.7.1 Criteria and standards for assessing the licensee's overall ageing management programme

2.233. The main criteria and standards for assessing the licensee's overall AMP are the IAEA Safety Guide NS-G-1.12, the WENRA Safety Reference Levels (SRL) Issue I (Ref 2) and ONR's Safety Assessment Principles (SAPs, Ref 3).

2.234. The ONR SAPs on ageing management, described in paragraph 2.18, are broadly similar to the SRLs Issue I for ageing management. Hence compliance with the ageing management SAPs and TAG-50 for ageing management has been used in this chapter for compliance with the SRL Issue I.

2.7.2 Use of International standards

2.235. Both licensees claim that their management systems ensure that ageing management is consistent with IAEA standards and WENRA reference levels. These standards have been incorporated into:

- EDF-NG's Nuclear Safety Principles (NSP) for AGRs and its Nuclear Safety Assessment Principles (NSAP) for Sizewell B
- NNB's Nuclear Safety Design Assessment Principles (NSDAP)

ONR's assessment confirms that these documents reflect the higher level ageing management requirements set in the international standards. More detailed requirements are discussed in the following two sections.

2.7.3 ONR's assessment of EDF-NG's overall ageing management programme

2.236. EDF-NG has noted that ageing management is integrated into its management system, but it does not have a specific ageing management process. ONR recognises this and its SAPs note that whilst a specific AMP may be used, it is not a requirement. ONR therefore agrees that the approach adopted by the licensee is acceptable, provided it results in suitable ageing management processes for all SSCs.

2.237. Figure 2.2 describes EDF-NG's key processes for ageing management and is presented in a format consistent with the systematic approach to ageing management described in IAEA Safety Guide NS-G-1.12. ONR considers that this is a suitable representation of the processes that control ageing management and it confirms that the licensee's AMP appears consistent with the IAEA Safety Guide.

2.238. The licensee has Maintain Design Integrity (MDI) and Equipment Reliability (ER) as the key processes for managing ageing with Technical Governance (TG) as a supporting process. ONR's assessment conclusions of the applicability of these three processes to ageing management is as follows:

- The key aspect of the MDI process is the production, maintenance and development of the safety case. It also includes the safety case anomalies process (SCAP), which is used to assess the implications of unexpected changes to the plant and can include plant degradation due to ageing. ONR's assessment and inspection processes include consideration of many aspects of MDI. ONR confirms that the MDI process includes appropriate consideration of ageing.
- The ER process is used to monitor the plant, to ensure that it performs as required and to give forewarning of potential plant failures. ONR uses the information from application of the ER process during its System Based Inspections (SBI) and can confirm through its own usage that it provides adequate monitoring of the effects of ageing for individual systems. For the TPR, ONR has inspected the programmes for past and future safety case health reviews and the effectiveness of the plant health committees and can confirm that these appear appropriate for monitoring the effects of ageing. Overall, ONR supports the licensee's claims for ER.
- The purpose of the TG process is to specify and monitor the engineering guidance and standards for the design and control of the plant. This includes two main types of document:
    - Company Technical Standards (CTS) which are mandatory;

- Technical Guidance Notes (TGN) which provide company guidance on how to comply with the CTS and are not mandatory

ONR has inspected the use of the TG process and can confirm that it appears to provide extensive coverage of the engineering systems and that the potential for ageing is included appropriately in EDF-NG's standards and guides.

2.239. As described in section 2.3.1.1, EDF-NG employs an extensive number of other processes to manage ageing at its sites. ONR's assessment of the key ones are in the following paragraphs (2.240 to 2.242).

2.240. Periodic Safety Review (PSR) provides a ten-yearly review of safety at the plants. ONR can confirm that ageing is explicitly included in the licensee's PSR process as set out in the IAEA's Safety Guide SSG-25 (Ref 20). In recent years, ONR has assessed the latest PSRs for Sizewell B (Ref 29) and Hinkley Point B and Hunterston B (Ref 30). In each case ONR concluded that the licensee had carried out an adequate PSR to justify operation for the following ten years. The latest PSRs for the other AGR stations have all been started and are due to be submitted to ONR by January 2019, with regulatory decisions by January 2020. ONR supports the licensee's claims for PSR.

2.241. EDF-NG's strategic approach to ageing is included in its Plant Life Extension (PLEX) projects, which cover the whole plant. ONR has reviewed submissions for irreplaceable components (Ref 31 to 33) and concluded that these did not reveal any significant flaws or omissions. ONR has further engaged with the licensee on its medium term plans for investment in the plants and was satisfied that the major investments it would expect to see for each station were present, including increased investment in corrosion, boiler modifications and nitrogen plant etc. ONR considers that the licensee is planning suitable investments in irreplaceable components to ensure that the effects of ageing are adequately mitigated.

2.242. The licensee's monitoring, testing, sampling and inspection activities are controlled through a maintenance schedule as required by the site licence and consist of:

- Routine activities during periods of plant operation. ONR inspects these on a regular basis, looking at the process as a whole and then specific systems during its SBIs
- Major activities during statutory outages once every three years for AGRs and refuelling outages every 18 months for the Sizewell B PWR. ONR inspects the licensee's activities during the outages and will only give consent to restart if it is content that work has been satisfactorily completed and the plant can be operated safely until the next outage.

ONR therefore has regular oversight of the licensee's monitoring, testing, sampling and inspection activities and can confirm that they are appropriate for ageing management.

2.243. Within its review of ageing management reported in this chapter, the licensee has identified two areas for improvement:

- The description of the EDF-NG ageing management arrangements, described in Chapter 2 of this report, should be issued in a formal company guidance document by 31 March 2018 (paragraph 2.223).
- The periodic review and update of TG documents should include review of relevant IGALL documents. EDF-NG has advised that the TG document

review management process will be updated by 28 February 2018 to include the requirement to review IGALL during the three-yearly cycle of TG document updates and hence this will be completed by mid-2021 (paragraph 2.222).

ONR accepts that these improvements are appropriate and that the dates for their completion are acceptable.

2.244. EDF-NG has a number of review processes embedded within its arrangements which contribute to understanding the impact of ageing on its plants. It has already been noted that ageing is not treated as a standalone process. Similarly, ONR notes that the licensee does not have a review process that looks at the impact of ageing as a whole. Given that the AGRs are reaching the ends of their operating lives, ONR considers that the licensee should ensure that it builds on its current processes for plant reviews and ensures that ageing management is explicitly considered. ONR has therefore asked the licensee to review its arrangements for the annual reporting of ageing management, and to include ageing management within an appropriate oversight process for each station for its reviews of performance for the calendar year 2018. These reviews for 2018 will start in early 2019, and hence the process will need to be in place by 31 December 2018.

2.245. In addition to its evaluations against the requirements of WENRA's technical specification for national reports, EDF-NG has illustrated its processes for ageing management by describing how it enacts its AMP for graphite cores and reactor internal components. These are the two key AGR plant areas which are subject to important ageing mechanisms and significant regulatory interest. ONR agrees that these sections are an accurate description of the ageing management processes for these plant areas.

2.246. From the assessment above, ONR concludes that EDF-NG has an adequate AMP. Nevertheless some secondary improvements have been identified in the present review that would be beneficial to ageing management and are now to be taken forward by EDF-NG to the timescales above, which it has agreed.

2.7.4 ONR's assessment of NNB's overall ageing management programme

2.247. Construction of the two reactors at Hinkley Point C by NNB started in March 2017. It is therefore much less advanced than EDF-NG in developing its AMP. Nevertheless, it has processes for incorporating ageing management into the design, primarily by assessing the impact of ageing. Though NNB has not yet started to develop monitoring programmes in any meaningful way, ONR would not expect it to at this stage.

2.248. The EPR™ design was subject to the UK's Generic Design Assessment (GDA), which is a non-site specific assessment of the design prior to the issue of a Design Acceptance Certificate. ONR's GDA reviewed the design against ONR's SAPs, including those associated with ageing. Reports of the final step, Step 4, of the GDA process are available on the ONR website (Ref 24). The GDA did not find any major issues with ageing management, although it did produce assessment findings in a limited number of areas that require the licensee to provide further justification of the impact of ageing on Hinkley Point C.

2.249. NNB has assessed the latest pre-construction safety case against its NSDAPs and found no gaps in its safety case. It has, however recognised that it needs to formalise its ageing management arrangements by producing a Corrosion and Ageing Management Strategy report in early 2018. ONR supports the production of this report.

2.250. ONR concludes that NNB has an adequate AMP for Hinkley Point C, given that it is still under construction.

2.7.5 Overall conclusions on the overall AMPs

2.251. The two licensees, EDF-NG and NNB, have different processes for ageing management. ONR accepts that these reflect the different lifetime stages for their stations.

2.252. For EDF-NG, the stations require a full AMP. The licensee has noted that whilst it does not have a specific ageing management process, it nevertheless has all the elements necessary within its management system. ONR notes that this is consistent with its regulatory expectations. ONR's assessment of the licensee's contribution to this chapter of the national assessment report demonstrates that the licensee has an adequate overall AMP.

2.253. Hinkley Point C is in the early stages of construction and hence the focus of ageing management is the design process and the "Understand and Plan" stages of the IAEA's ageing management model. ONR considers that this is appropriate and that NNB has an adequate overall AMP for this stage of its project.

2.254. In spite of having adequate AMPs, ONR's assessment has identified a number of areas for both licensees where improvements to ageing management would be beneficial. For the overall AMP, these are as follows:

- EDF-NG should issue a formal company guidance document to describe ageing management arrangements, described in Chapter 2 of this report, by 31 March 2018 (paragraph 2.243).
- EDF-NG should include review against relevant IGALL documents in the periodic review and update of TG. EDF-NG has advised that the TG document review management process will be updated by 28 February 2018 to include the requirement to review against IGALL during the three-yearly cycle of TG document updates and hence this will be completed by mid-2021 (paragraph 2.243).
- EDF-NG should review its arrangements for the annual reporting of ageing management, and to include ageing management within an appropriate oversight process for each station for its reviews of performance for the year 2018, which will start in early 2019. (paragraph 2.244).
- NNB should formalise its ageing management arrangements by producing a Corrosion and Ageing Management Strategy report by 30 June 2018 (paragraph 2.249).

2.255. To ensure that these improvements are implemented in a timely manner they have been brought together along with those from other chapters in Chapter 9. Chapter 9 identifies a single area for improvement (AFI) for each licensee to undertake a programme of improvement and the individual elements necessary to complete that programme are clearly identified, along with dates for their completion.

2.256. In summary, the main conclusion from ONR's assessment is that both licensees have adequate AMPs, given the specific stages of their lifecycles, but that both need to make a limited number of secondary beneficial improvements.

3 Electrical cables

3.1 Description of ageing management programmes for electrical cables

3.1.1 Scope of ageing management for electrical cables

3.1.1.1 Operating reactors (EDF-NG)

3.1. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

3.2. Whereas Section 2 of this report describes EDF-NG's general arrangements available for the ageing management of all SSCs, this section considers the ageing management programmes for electrical cables, a component within numerous systems across the station. The general ageing management arrangements across all systems affected are discussed in Section 2, and will not be repeated here. This section will instead focus on the specific arrangements for the grouping of safety-related electrical cables.

3.3. It is evident from Section 2 that the safety case (managed by the MDI process) is fundamental to the ageing management of safety-related SSCs. The construct of the safety case is typically at plant or system level, covering normal operation faults and hazards. Each safety case identifies the safety related SSCs supporting the safety case, and their ageing management arrangements, including the overall maintenance, testing and inspection schedule.

3.4. ER is used by EDF-NG to manage the day to day plant condition and performance in accordance with the safety case. As noted in Section 2 the focus of ER is at system level, and predominantly upon plant and system components where active change in plant configuration and performance can be readily measured and managed. ER is applied to all systems containing safety related electrical cabling.

3.5. TG is used by EDF-NG to provide additional support to the safety case and to ER, in specifying how the plant is designed and controlled. TG is based around disciplines, groups of systems, structures or components, which provide an effective means of capturing and disseminating discipline based national and international standards, and imparting knowledge from discipline based Chief Engineers. Safety related electrical cables are designed and controlled using TG and hence the focus of the commentary in Section 3 is on the TG arrangements for the ageing management of electrical cables.

3.6. Ageing of electrical cables in NPPs can occur as a result of thermally induced effects, irradiation induced effects or due to adverse environmental conditions (e.g. excessive moisture). These ageing factors can act individually or in combination depending on the function and location of the cable concerned. For example, the most significant ageing factor for power cables is likely to be thermal, whereas for instrumentation and control cables any combination of the three ageing factors may be present. Furthermore, the ageing effect of combined radiation and thermal exposure is not fully understood. For power cables, the insulation is normally the component of cable construction that is most vulnerable to degradation with age. In the widely used polymeric insulation materials, ageing causes embrittlement of the insulation leading to cracking and eventual electrical breakdown. The extent to which the occurrence of cracking precedes electrical breakdown is not understood precisely but varies according to insulation type. For control cables the environment dominates and the outer sheath is the most vulnerable component.

3.7. Historically, monitoring of cable ageing was carried out as part of routine maintenance and inspection activities or during replacement of defective plant. For example, insulation resistance (IR) and polarisation index (PI) checks carried out during plant maintenance when trended were deemed to be an effective method of establishing cable deterioration. The maintenance and defective plant replacement routines also provided the opportunity for visual inspection of many thousands of cables. This approach was successful for many years and no generic problems manifested themselves.

3.8. However, in the last ten years it was realised that with the AGRs approaching their design lives this approach was no longer sustainable and that a systematic inspection and monitoring programme should be established. In support of the ageing management of cables a Technical Guidance Note (which is part of the TG process, paragraph 2.82) on cable condition monitoring has been established and implemented, with details given below.

3.9. The Technical Guidance Note provides the basis for understanding cable ageing issues, and assists the inspector in determining when and where cable ageing is occurring and suggests appropriate remedial action.

3.10. For the Water Cooled Reactors, an extensive qualification programme has been followed to ensure the cables meet specific performance requirements and these include induced ageing tests.

Cables considered under the ageing management programme

High Voltage Cables Subject to Adverse Environment

(i) <u>Paper Cables (≥11kV)</u>

3.11. EDF-NG's general experience with paper cables is that, given good quality design, manufacture and operation within its design envelope the cables should have a life expectancy of the order of 60 to 80 years. Poor quality control during manufacture (a rogue batch) or poor installation technique may however reduce this anticipated lifetime. Cable support structures and armouring may also degrade with time. The occurrence of such lifetime threats is believed to be relatively infrequent in the relatively mild environment of power station cable flats and raceways.

3.12. EDF-NG notes that disturbing or moving old paper insulated, lead sheathed cables will often lead to failure of the cable. This is due to voids being created between the lead sheath and the cable cores. The general advice promulgated within the company is to avoid disturbing old cables.

3.13. EDF-NG technical guidance provided to the stations recommends that two cable routes be subject to visual/instrumented inspection.

(ii) <u>PE, XLPE and EPR Polymeric Cables (≥3.3kV)</u>

3.14. These polymeric insulation types have largely been used for high voltage single core cables on the more recent AGR power stations. The performance of these cables within the company and worldwide has been variable and life prediction techniques difficult to apply, as the dominant ageing mechanisms may be different from one cable design to another.

3.15. EDF-NG considers that the weakest components in PE (Polyethylene), EPR (Ethylene Propylene Rubber) and XLPE (cross-linked Polyethylene) cable circuits are the cable joints and terminations. These items are susceptible to moisture, thermo-mechanical movement and external mechanical damage. The general advice promulgated within the company is not to move a cable near to a joint or termination position.

3.16. Early designs of polymeric cables may be subject to “treeing” phenomena which can ultimately lead to insulation breakdown of the cable. Treeing is an electrical pre-breakdown phenomenon in solid insulation. It is a damaging process due to partial discharges and progresses through the stressed dielectric insulation, in a path resembling the branches of a tree. In the event of a failed cable it is recommended that microscopic examination is undertaken to assess the effect and extent, if any, of “treeing”.

3.17. EDF-NG’s technical guidance provided to the stations recommends that visual inspection of five representative cable routes should be undertaken at six yearly intervals.

Medium voltage cables buried or in trenches

(iii) PVC Cables (≥415V)

3.18. The significant ageing factor for PVC power cables is considered to be thermal.

3.19. The majority of power cables are not operated at their maximum design current rating, as in most cases voltage drop rather than current carrying capacity is the overriding factor when sizing cables. For this reason cables generally operate at temperatures considerably less than their maximum design operating temperature, therefore increasing the predicted lifetime of the cable insulation.

3.20. Distribution feeder cables may have undergone an increase in load over the years of operation, due to system extensions/modifications. Therefore circuits of this nature are more likely to have continuously loaded cables operating near their design ratings, making their lifetime less certain. EDF-NG recommends the use of a thermal image camera to record the operating temperature of such cables during walk downs.

3.21. It is known that if continually operated at its thermal rating PVC insulation may become hard and brittle after approximately 30 years. The extent to which the embrittlement and subsequent cracking may occur before electrical breakdown will occur is not precisely known and can vary from one cable polymer compound to another.

3.22. EDF-NG advises that the above identified cables are visually inspected and if deemed necessary additional condition assessments using non-intrusive techniques i.e. electrical tests, chemical micro-sampling or mechanical (e.g. cable indenter), are undertaken. If the assessment indicates that the insulation of these cables has degraded significantly, highlighted by either trending or a point measurement, then the cable is earmarked for a sample to be removed and sent to a cables specialist for residual life assessment analysis. Depending on the outcome of the life assessment, the result will either confirm that no further detailed assessment is necessary or recommend further testing.

3.23. EDF-NG technical guidance provided to the stations recommends that approximately ten cables are selected for visual/tactile inspection of which at least two will be instrumentation and control cables. The remainder will be a representative range of power cables selected in accordance with the guidance note.

(iv) Mineral Insulation Covered Cables (MICC) (≤415V)

3.24. Mineral insulation does not age in the same sense as organic materials and is unlikely to deteriorate if kept moisture free. MICC cable termination sleeves are made of organic materials. EDF-NG recommends that approximately five cables should be visually inspected for evidence of embrittlement. If the sleeving is found to be brittle, a new seal should be fitted and assessment of all remaining cables determined. In addition, the two cables judged to be the most degraded, should be disconnected from the associated equipment and checked for insulation resistance.

Neutron Flux Instrumentation Cables

(v) <u>Special Cables</u>

3.25. These cables are those whose construction has insulation or sheathing consisting of polymeric materials different from those above. The cables most at risk are those subject to radiation ageing with insulation consisting of PEEK (Polyether ether ketone), Kapton (polyimide), XLPA (cross linked polyalkene) and SR (silicone rubber). The cables are typically used on the reactor pile cap or within fuel route equipment for control and indication purposes. During normal operation and under reactor fault conditions such cables are likely to be subject to radiation exposure in combination with thermal ageing effects. EDF-NG's guidelines recommend that visual inspection of five representative cables of each type is undertaken at six yearly intervals. Should reactor faults occur giving high radiation exposure rates to a group of cables, visual inspection of the exposed cables following the fault is recommended, to verify that the insulation and sheathing materials are of adequate strength.

3.1.1.2 Hinkley Point C (NNB)

3.26. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

3.27. For HPC, NNB advises that an extensive cable qualification programme will be followed to ensure the cables meet specific performance requirements and these include induced ageing tests. A new family of cables will be qualified for the EPR^{TM} based on those used at Flamanville 3 (FA3). Additional qualification testing, where required, will be added to cater for cables in a UK context. Then, throughout the stations life, in line with an AMP, ongoing inspections, monitoring and where necessary testing will be undertaken to add confidence that the ageing process is following an anticipated path.

3.28. In addition to the inspection of the normal cable installation which comprises conventional containment systems constructed of trays and ladders additional appropriate inspections and monitoring will be considered to cater for functional or fire load reduction cable tray wrapping. Functional cable tray wrapping is installed to protect selected cabled circuits from common cause failure and fire load reduction cable tray wrapping is installed to take the wrapped cable out of the equation for combustible mass in the fire compartment. They are discussed further in later sections of this report.

3.29. NNB notes that procedures and criteria will be prepared and cables or cable groups will be selected for monitoring and trending of the ageing process. This will apply to all parts of the cable installation including conductors, insulation, armour (where applicable) screens, sheaths and termination arrangements etc.

3.30. Internal and external OPEX will be incorporated into the development of the Ageing Management Programme (AMP).

3.31. Where necessary external R&D facilities may be employed if such analysis is required. That and any other external relevant bodies programmes will be taken into consideration,

3.1.2 Ageing assessment of electrical cables

3.1.2.1 Operating reactors (EDF-NG)

Identification of cables most likely to age prematurely

3.32. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

3.33. When EDF-NGL identifies cables for its inspection, due consideration is given to locations susceptible to localised environmental conditions and the selection criteria identified in the guidance note. Cables age much more quickly than would of otherwise have been expected when the following conditions occur in the vicinity of cables:

- Elevated surface temperature of insulated pipes.
- Missing pipe insulation or gaps in the insulation.
- Valve operators and other devices on steam or hot process lines.
- Continuously energised solenoid operated valves.
- Localised high temperature areas.
- Fire blankets/enclosures installed post cable design.
- Areas exposed to unusual lighting (e.g. direct sunlight or close proximity to artificial lighting).
- Near moving or vibrating equipment.
- Near fluid systems that tend to leak.
- Radiation hotspots.
- Manipulation of aged cables at terminations.

3.34. The stations' Health Physics department are able to identify thermal and radiation hot spot areas across the plant. Discussions with operations and maintenance engineers, and identification of previously failed cables, assists in identifying areas or cables of concern.

Causes of cable ageing

3.35. The effects of severe ageing degradation of cable materials can include embrittlement, cracking or crazing, discolouration, melting, and a change in the mechanical and electrical properties that are essential for the cable to perform its design function. Ageing of cables is predominantly caused by exposure to the following stressors:

(i) <u>Heat</u>

3.36. Thermal ageing results from the exposure of cable materials to normal and abnormal thermal environments. The normal ambient room temperature in most plant areas results in very slow degradation of cable insulation and jacket materials. However, localised elevated temperature or radiant heat from sources such as process lines that are too close, degraded/inadequate lagging or inadequate ventilation can produce severe damage relatively rapidly, with the damage generally limited to one section of the cable. In these instance efforts have been made to divert cables or where this has not been feasible install heat shields. Ageing of the insulation can also result from ohmic heating of cable conductors, if an undersized conductor is used or the combination of ambient temperature/environment and ohmic heating is not properly considered during cable sizing. A typical example being the installation of fire blankets around cables, to create a segregated route, post cable design.

3.37. Hot spots affect only limited portions, if any, of a given cable run. Nonetheless, they are a major concern because the rate of material degradation within the hot spot can be significantly higher than that occurring elsewhere.

3.38. Ohmic (I^2R) heating, resulting from the electrical current in power cables, generally affects the entire length of a cable run. The higher the current in relation to the cable design rating, the greater will be the rate of heat generation. Thermal ageing that results from conductor ohmic heating is significant in power cable applications where the connected load is operating for a significant percentage of its installed lifetime, and the current during such operation is a substantial percentage of the cable design rating.

(ii) Radiation

3.39. The radiation tolerance of an individual material will vary based on the general type of material and its chemical structure and formulation. Organic materials commonly used in nuclear plant cable applications vary widely in their radiation resistance. Changes in the materials overall mechanical properties (such as elongation at break, tensile strength and hardness) and electrical properties (such as dielectric strength and conductivity) can result from exposure to radiation.

3.40. Tables 3.1 and 3.2 are used as a guide to radiation limits for various cable types.

(iii) Manipulation

3.41. Wear ageing of cable system components pertain only to terminations and/or connections and results primarily from manipulation (bending) of the component during maintenance or testing. If cable jackets or insulation are found to be hardened or resistant to flexing, they should not be bent further. Problems with cracked plumb joints at motor cable boxes are not unknown and this is, in most part, due to not providing adequate support arrangements to the cables and cable box when exchanging HV motors, such as the Cooling Water Pump motors.

3.42. A summary of all commonly known ageing stressors, observed ageing mechanisms, degradations, and potential effects of ageing on the components of a low voltage cable system is presented in Table 3.3 below.

Polymeric cable insulation (Note 1)	Maximum tolerable dose (Note 2)
Polyimide (Kapton)	10^7 Gy (10^9 Rad) Strength to 50% by 5 x 10^7 Gy
PEEK	10^7 Gy (10^9 Rad) Extremely resistant to embrittlement
Silicone Rubber	10^4 Gy (10^6 Rad) Should not use above 10^5 Gy Becomes very brittle
PVC	10^6 Gy (10^8 Rad)

Polymeric cable insulation (Note 1)	Maximum tolerable dose (Note 2)
EPR	10^6 - 10^7 Gy (10^8 – 10^9 Rad) depending on curing process
PVDF	3 x 10^5 Gy (3 x 10^7 Rad) embrittles
CSP (Hyperlon)	10^5 Gy (10^7 Rad) Can liberate HCl, SO_2 gas with high dose (Note 3)

Table 3.1 A guide to insulation total dose limits (Gamma)

Polymeric cable insulation	Maximum tolerable dose (Note 2)
Normal operation	10^4 Gy (10^6 Rad) over 30 yrs
Stuck fuel faults	10^4 Gy/hr (10^6 Rad/h) over 2½ hr Assume dose absorbed as 10^5 Gy for the single event

Table 3.2 Pile cap cable Gamma radiation exposure

Note 1. These vs are associated with a general reduction in mechanical properties by about 25 % of their original value.

2. Some materials are dose rate dependent.

3. Use of halogen based insulating materials and compounds in a γ radiation environment is best avoided.

Component	Sub-Component	Applicable Stressors	Ageing Mechanisms	Degradation	Potential Effects
Cable	Insulation and jacket	Heat, oxygen	Thermal and thermoxidative degradation of organics from ambient and ohmic heating.	Embrittlement, cracking, melting, discolouration.	Reduced insulation resistance (IR); electrical failure; increased vulnerability to failure in harsh environments.
		Radiation, oxygen	Chemical decomposition of organics; radiation-induced oxidation.	Embrittlement, cracking, discolouration, swelling.	Reduced IR; electrical failure.
		External mechanical stresses	Wear resulting from work in area, personnel traffic, or poor support practices.	Cuts, cracking, abrasion, tearing.	Reduced IR; electrical failure.
Connector	Contact surfaces	Electrochemical stresses (moisture, oxygen)	Corrosion and oxidation of metals.	Corrosion and oxidation of external surfaces of contacts.	Increased resistance and heating; loss of circuit continuity.
Compression fitting	Lug	Vibration, tensile stress	Deformation and fatigue of metals.	Loosening of lug on conductor; breakage of lug.	Loss of circuit continuity; high resistance.

Table 3.3 Summary of stressors, significant and observed ageing mechanisms, degradations and potential effects.

Acceptance criteria related to the ageing mechanisms

Cables visually examined

3.43. It is recommended that all observations recorded during the visual examination be tabulated for each cable. This should enable a general assessment of condition to be made. With some experience an assessment of condition may be made from the 'feel' of the insulation such as hardness (leading to embrittlement) particularly if slightly bent to see if there is any significant increase in stiffness. If all the cables and cores are found to be in a similar condition the assessment is straightforward, but will be more difficult if this is found not to be the case. There may be some variation between cable manufacturers if more than one manufacturer is involved. Seemingly isolated examples of degradation may need to be further examined to see if this is localised and can be attributed to another cause such as a nearby source of heat, etc. and separate action taken. If a common pattern cannot be identified the size of the sample examined will need to be increased until such a pattern can be established. It should then be possible to fit the assessment of each cable type into one of the following categories.

i) No evidence of ageing degradation

3.44. If the cable type shows no evidence of ageing degradation it is considered that the cable type is suitable for a further six years' service.

ii) Limited, but not significant, ageing degradation

3.45. This would include observations such as mild discolouration, some evidence of deterioration of mechanical properties e.g. insulation feels harder and noticeably stiffer if slightly bent, the presence of limited surface crazing, etc..

3.46. It is not considered necessary to change any cables provided their condition is periodically monitored. It is recommended that a selection of cables showing limited signs of ageing be inspected for further change at yearly intervals.

iii) Limited significant evidence of ageing degradation

3.47. On the basis that electrical failure follows mechanical failure, significant ageing degradation should be taken to mean major loss of mechanical strength (approaching a condition where mechanical failure, i.e. cracking of insulation could occur resulting in loss of dielectric strength).

3.48. This would include observations ranging from the presence of more extensive surface crazing, hardness approaching embrittlement, complete loss of mechanical strength i.e. cracks when slightly bent, closed cracks, open cracks (including polluted cracks indicating that the cracks had been present for some time), perished, etc..

3.49. If significant evidence of ageing degradation is found on a limited number of cables, then all the cables of this cable type need to be visually examined.

3.50. If difficulty is experienced in ascertaining the actual condition it may be appropriate to obtain a test sample, than can be sent for further analysis. Those cables showing significant signs of ageing should be replaced with a selected number of the remaining cables monitored at yearly intervals.

3.51. For mineral insulated cable this only applies to the termination.

Electrical testing of cables

i) Insulation Resistance

3.52. Generally, cables supplying motors and transformers have Insulation Resistance and Polarisation Index (IR/PI) tests undertaken, as part of routine maintenance. To take cognisance of this testing a copy of the IR/PI results are held within the cable inspection schedule.

3.53. The cable inspection schedule is maintained within an appropriate Engineering database. Maintenance of the cable inspection schedule is the responsibility of the Stations' Electrical Engineering Group.

3.54. The insulation resistance of multicore cables shall not be less than 10M.ohms.km between cores and between cores and armour, and not less than 10k.ohms.km between armour and earth at 20°C.

ii) Partial Discharge testing

3.55. Partial Discharge (PD) is a localised electrical discharge, due to inability of the insulation to withstand the local electrical stress. PD is initiated by a strong electric field and is governed by applied voltage. When the applied voltage is such that it generates an electrical stress greater than local insulation can withstand, breakdown occurs.

3.56. The charge transfer, which occurs in time frames of nano-seconds, generates heat, light and chemical change, and it is these by-products which act as stress enhancers and accelerates the deterioration of the insulation.

3.57. The characteristics of the electrical signal generated by the PD (magnitude, pulse shape, repletion rate, etc.) are influenced by the local degradation conditions. Thus, PD detection and characterization can, in turn, provide information on the location, nature, form, and extent of degradation. As a result, PD monitoring has increasingly become an important part of condition monitoring in utilities.

3.58. The action levels for PD activity within HV cables is shown in Table 3.4.

Discharge level	Solid insulated MV Switchgear	Air insulated Switchgear	PILC and MV cable accessories	MV EPR cables	MV XLPE cables
1	0—15dB	0—10dB	0—250pC	0—50pC	0—30pC
2	15—25dB	10—15dB	250—500pC	50—120pC	30—80pC
3	25—35dB	15—30dB	500—1500pC	120—250pC	80—150pC
4	>35dB	>30dB	>1500pC	>250pC	>150pC

Table 3.4 Action levels for PD activity within HV cables

Key standards and guidance

3.59. The Technical Guidance Note (paragraph 3.8 onwards) provides the basis for understanding cable ageing issues, and assists the inspector in determining when and where cable ageing is occurring and suggests appropriate remedial action. In producing the Technical Guidance Note the following documents were reviewed:

- EPRI – 1020804 - Ageing Management Program Development Guidance for AC and DC Low- Voltage Power Cable Systems for Nuclear Power Plants, June 2010.
- EPRI – 1020805 - Ageing Management Program Guidance for Medium-Voltage Cable Systems for Nuclear Power Plants, June 2010.

- EPRI – 1021629 - Plant Support Engineering: Ageing Management Program Development Guidance for Instrument and Control Cable Systems for Nuclear Power Plants, November 2010.

3.60. IAEA also produce a programme of International Generic Ageing Lessons Learned (IGALL) reports for Nuclear Power Plants. These reports supplement the IAEA Safety Reports Series No. 82, Ageing Management for Nuclear Power Plants. For structures, systems and components important for safety, the following information is presented:

- a generic sample of ageing management review tables;
- a collection of proven ageing management programmes; and,
- a collection of typical time limited ageing analyses.

3.61. The IGALL documents relevant to the ageing management of cables are listed below:

- AMP201 Insulation Material for Electrical Cables and Connections not Subject to Environmental Qualification Requirements.
- AMP202 Insulation Material for Electrical Cables and Connections not Subject to Environmental Qualification Requirements Used in Instrumentation Circuits.
- AMP203 Inaccessible Power Cables not Subject to Environmental Qualification Requirements.

3.62. These safety reports describe practices and techniques for the inspection, mitigation of ageing degradation, corrective action including repair methods, and operating experience for cables. It also provides general guidance for developing an effective ageing management process for cables based upon the following principles:

- understanding ageing mechanisms;
- detection of ageing effects;
- monitoring and trending of ageing effects;
- mitigating ageing effects;
- acceptance criteria; and,
- corrective actions.

3.63. The IAEA guidance provides specific information for the ageing management of cables. It is judged that the EDF-NG procedures for ageing management align with the intent of the IAEA guidance and there are no significant gaps identified.

Procurement Specifications

3.64. EDF-NG's internal standard specifies the design, performance, acceptance and/or in-situ test limits and reliability requirements, as appropriate when procuring cabling. These standards ensure that known cable degradation affects do not prematurely manifest themselves, and include the following:

BS EN 10111	Continuously hot rolled low carbon steel sheet and strip for cold forming.
BS EN 1449-1.1	Steel plate, sheet and strip. Carbon and carbon-manganese plate, sheet and strip. General specification.

BS EN 1461	Hot dip galvanized coatings on fabricated iron and steel articles. Specifications and test methods.
BS 3858	Specification for binding and identification sleeves for use on cables and wires.
BS 4320	Specification for metal washers for general engineering. Metric series.
BS EN 50288-7	Multi-element metallic cables used in analogue and digital communication and control. Sectional specification for instrumentation and control cables.
BS EN 50525-1	Electric cables – Low voltage energy cables of rated voltages up to and including 450/750V (U_0/U).
BS 5467	Electric cables. Thermosetting insulated, armoured cables for voltages of 600/1000 V and 1900/3300 V.
BS 6004	Electric cables. PVC insulated and PVC sheathed cables for voltages up to and including 300/500 V, for electric power, lighting.
BS EN 60079-0	Explosive Atmospheres - Equipment - General Requirements.
BS EN 60079-14	Explosive Atmospheres – Electrical installations design, selection and erection.
BS EN 60332-3-22	Tests on electric and optical fibre cables under fire conditions. Test for vertical flame spread of vertically-mounted bunched wires or cables.
BS EN 60352-2	Crimped connections – General requirements, test methods and practical guidance.
BS EN 60702-1	Mineral insulated cables and their terminations with a rated voltage not exceeding 750V.
BS 6121-1	Specification for metallic cable glands.
BS EN 61238-1	Compression and mechanical connectors for power cables for rated voltages up to 36 kV (Um = 42 kV). Test methods and requirements.
BS 6207-3	Mineral insulated cables with a rated voltage not exceeding 750 V. Guide to use.
BS 6387	Specification for performance requirements for cables required to maintain circuit integrity under fire conditions.
BS 6622	Electric cables. Armoured cables with thermosetting insulation for rated voltages from 3.8/6.6 kV to 19/33 kV. Requirements and test methods.
BS 6724	Electric cables. Thermosetting insulated, armoured cables for voltages of 600/1000 V and 1900/3300 V, having low emission of smoke and corrosive gases when affected by fire.
BS 7211	Electric cables. Thermosetting insulated, non-armoured cables for voltages up to and including 450/750 V, for electric power, lighting and internal wiring, and having low

	emission of smoke and corrosive gases when affected by fire.
BS 7671	Requirements for electrical installations. IEE Wiring Regulations. Seventeenth Edition.
BS 7835	Electric cables. Armoured cables with thermosetting insulation for rated voltages from 3.8/6.6 kV to 19/33 kV having low emission of smoke and corrosive gases when affected by fire.
BS 7846	Electric cables, Thermosetting insulated, armoured, fire resistant cables of rated voltage 600/1000V, having low emission of smoke and corrosive gases when affected by fire – specification.
C89	Performance specification for terminations on polymeric insulated cables rated at 12kV and 36kV maximum system voltage.
IEC 60502	Power cables with extruded insulation and their accessories for rated voltages from 1kV up to 30kV.

R&D programmes

(i) Online Partial Discharge monitoring

3.65. As described in Section 3.1.2.1 partial discharge testing provides information on the location, nature, form, and extent of cable degradation and is increasingly becoming an important part of condition monitoring in utilities. Although online Partial Discharge monitoring has been widely used for cables, difficulties in the interpretation of measurement results (location and criticality) remained to be solved. It was for this reason that in collaboration with Caledonian University in Glasgow EDF-NG conducted a research programme whose main objectives were as follows:

- to develop further knowledge to interpret PD signal propagation in cable networks and to develop a portable PD monitoring system; and,
- to develop techniques for de-noising and PD source localisation.

3.66. The research and development work has been completed and the results have been very encouraging. The equipment will be shared across the fleet with an expectation that all HV single core cables are routinely monitored once per year for signs of partial discharge activity.

3.67. Due to the sophisticated PD recognition and de-noising capability this equipment offers EDF-NG are currently exploring commercial opportunities with Caledonian University.

(ii) Development of a novel online cable IR measurement technique

3.68. In collaboration with Caledonian University the project aims to develop an on-line technique for cable IR testing, in an attempt to tackle the lack of knowledge of cable IR condition during the intervals between off-line tests. The project objectives include:-

- Theoretical analysis of leakage current in relation to cable length, equipment connected, and in relation to deterioration or ageing. Numerical simulation (software package: COMSOL) will be carried out to consider the thermal effects on insulation resistance and capacitance.

- Laboratory investigation of changes in insulation resistance and in capacitance in a range of cable types, under long term thermal stress and mechanical stress. The results would provide input to the COMSOL investigation.
- Determination of components of currents due to electro-magnetic induction and due to leakage, through numerical simulation and lab experiments.
- Analysis of the relationship between the harmonic components of the leakage current, relative to the fundamental component, and insulation deterioration.
- Study of signatures of incipient fault signals.
- The determination of criteria and measuring ranges for assessment of insulation condition.
- Development of instrumentation and software package for insulation condition monitoring via leakage current measurement.
- Developing an online testing technique for determination of earthing resistance and the relative change over time.
- Validation of the assessment criteria by carrying out leakage current measurement over a period of 2 years and comparing with off-line IR measurement results.
- And finally, investigate the source of background noise and harmonic contents across the power stations to be visited.

3.69. The deliverables of this project are:-

- A portable, on-line cable insulation condition monitoring system with bespoke software packages containing functions for data acquisition, data denoising, data management and insulation condition diagnosis and fault localisation, through a combination of IR, PD and incipient fault detection.
- Integration of the cable PD monitoring, motor monitoring techniques with the new techniques to be developed un the current project. This will help to improve the accuracy of online IR testing.

(iii) <u>High Voltage cable lifetime management</u>

3.70. Through initial discussions between Bruce Power NPP and The University of Strathclyde, the opportunity to develop more informative lifetime assessment capabilities as well as potential predictive on-line cable condition monitoring systems has been considered.

3.71. Two opportunities for project developments have been initially identified. Firstly, the application of lifetime prognostic/predictive analysis methods on legacy and current data captured over a number of years on different cable types in order to ascertain cable lifetime assessment as well as possible cable fault determination. Secondly, the opportunity to consider multi-parameter on-line condition monitoring techniques for cable integrity to permit continuous health monitoring assessment and to build up a database of information from which cable health index determinations, health alarms, historical trends and lifetime estimations may be evaluated to assist in maintenance decisions. The database would also permit application of machine learning methods for more robust large scale predictive assessment methods, anomalous event detection and characterisation, and provide large scale data sets for cable type/family comparisons and relevant business driver decisions.

3.72. The main research project objectives are as follows:-

- To understand and assess the current legacy time based monitoring data on HV and MV cables to initially understand the nature of the measurements and appreciate the current health assessment of the cables based on associated ageing profiles published by cable manufacturers.
- To understand the typical time dependent load nature on the cables.

3.73. The proposed project will focus on the following key deliverables:-

- Provide a framework which will enable assessment of current cable health from legacy cable measurements.
- Provide health trending and predictive analytics for cable health assessment from legacy cable measurements and continual off-line measurements.
- Provide a strategy for improved future cable measurement and assessment implementation (e.g. continuous online and off-line measurements e.g. PD measurements).
- Produce a technical implementation specification for future cable measurements, trending analytics.
- Develop a set of cable asset management health indicators to improve asset management.

3.1.2.2 Hinkley Point C (NNB)

3.74. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

3.75. For the EPR^{TM} the cable specification is based on the AFCEN (French Society for Design and Construction Rules for Nuclear Island Components) Code for Design and Construction Rules for Electrical Components of Nuclear Islands (RCC-E) 2012. To cater for UK context the following is also used, the UK EPR^{TM} Book of Project Data Supplementary to Requirements of RCC-E 2012. These together with internal technical specifications form the procurement specification for cables for use in the UK EPR^{TM}.

3.76. Over the course of the project life cycle these appropriate design documents along with manufacturing, test, commissioning and operational documents will be used in the development of our Ageing Management (AM) strategy.

3.77. The monitoring, testing, sampling, and inspection activities performed internally and by third party organisations will be developed. These will include the description of activities, frequency of them and acceptance criteria, how programmes are implemented, how trending is investigated and how unexpected degrading mechanisms are identified.

3.1.3 Monitoring, testing, sampling and inspection activities for electrical cables

3.1.3.1 Operating reactors (EDF-NG)

3.78. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

Targeting cable routes for inspection

3.79. It is not considered practical or necessary to monitor the ageing factors of all cables/cable routes on a continuous basis. By targeting specific cables according to their importance, utilisation and environment for each generic type it is possible to make judgements on the condition of the remaining inventory of that type. This is particularly so if the targeted sample includes predicted worst case degradation.

3.80. As cables and cable routes are carefully targeted dependent on environmental factors, electrical loading, cable tray utilisation and importance it is possible to inspect a bounding case subset of cable / cable routes which may at a given station vary from between 2 to 10.

3.81. The purpose of the selection process is to define a representative number of cable routes whereby following inspection, assessment of operational data and where necessary specific testing, the general condition of all the cables of that type can be ascertained.

3.82. A short list of cables for inspection includes cables whose failure could result in:

(i) Potential nuclear safety consequences.

(ii) Commercial risk through loss of generation.

Typically cables circuits for inspection includes, but not limited to, cables associated with essential supplies, essential control and indication, back-up generation supplies, and high voltage transformer feeds and interconnectors.

3.83. The selection of cables gives priority to those which:

(i) are most heavily loaded with respect to their design rating; and,

(ii) have been subjected to environmental conditions which could lead to premature degradation of the cables on part or all of the cable route.

3.84. There will be cables that are located within areas that are generally considered as inaccessible during normal station business, i.e. trenches. Trench covers are removed for the purposes of pipework and civil inspections. Where possible, cable inspection routines are co-ordinated to align with these pipework and civil inspection routines to enable cables, associated cable support steelwork and accessories, and earth bars within these trenches to be inspected. The inspection and findings are recorded within the cable inspection schedule.

3.85. In instances where co-ordination with the pipework and civil inspection routines is not possible or cables are routed within dedicated cable trenches, then routines are generated for a systematic inspection of sample trenches. The selection of sample trenches for inclusion is based on the above cable targeting criteria, and is included in the cable inspection schedule.

Cable inspection schedule

3.86. A cable inspection schedule unique to each station has been produced. It identifies those cables/cable routes which have been targeted for inspection by a combination of raceway location, cable number and/or cable steelwork node numbers. The latter numbers, where available, uniquely identify the route in a given cable race. Consideration is given to include drawings showing general walk down areas.

3.87. Where standard System Health (ER) walk down check sheets are available, and appropriate to record the walk down findings, they are used. Where no suitable walk down check sheet is available, or individual cables of concern identified, the findings are recorded on the check sheet shown in Figure 3.1.

3.88. Generally, cables supplying motors and transformers have Insulation Resistance and Polarisation Index (IR/PI) tests undertaken, as part of routine maintenance. To take cognisance of this testing a copy of the IR/PI results are held within the cable inspection schedule.

3.89. The cable inspection schedule is maintained within an appropriate Engineering database. Maintenance of the cable inspection schedule is the responsibility of the Stations' Electrical Engineering Group.

Principles of cable inspection and testing

3.90. EDF-NGL assessment of cables that are prone to early ageing, and timely replacement or repair of degraded cables, will minimise unnecessary power losses, unplanned shutdowns, and challenges to the plants' safety systems.

3.91. The cable insulation and jacket materials are the focus of ageing management, with the ageing of the insulation being the most significant factor. In general, as the insulations and jackets age, they can become:

- Harder or softer
- Shiny or dull
- Change colour or hue
- Crack or craze

3.92. An infrared thermography camera is not a condition monitoring technique in itself, but it can be used to identify areas and describe the thermal ageing influences on the cabling system.

3.93. The most important part of any cable condition monitoring program is the walk down inspection. The inspection by EDF-NGL should focus on plant areas with adverse environments and identify the environmental conditions in the immediate vicinity of the cables.

3.94. A walk down with visual/tactile inspections of cables identified using the guidance is performed by EDF-NGL, so that an evaluation can be made as to whether or not significant ageing degradation is occurring and whether corrective actions need to be taken. When performing the inspection, inspectors are looking for, signs of degradation and installations that could cause adverse localised environments. In addition documenting of the plant status, environment conditions and date and time of the inspection should be noted. The guidance note describes causes of known cable ageing problems and contains immediate and long term recommended actions. It should be noted for those cables run in trunking the benefits of visual inspecting them are outweighed by the risk of damaging the outer sheath of the cables therein when removing / replacing the covers. For this small subset of cables, mostly found in the older AGR stations, reliance is placed on IR/ PI checks.

3.95. EDF-NGL inspectors should focus on detecting any visible change to equipment condition from the previous inspection. Degradation can appear as crazing, cracking, fading, textural changes or discolouration of materials. Unusual noises, deposits of foreign substances, stains from system leaks, and corrosion may also be indicators of adverse localised environments.

Figure 3.1
Cable inspection checksheet

CABLE HISTORY AND PRESENT CONDITION

Date of Inspection …………………Designation……Essential / Non-Essential

Cable No ………………………………………… Voltage Class …………………

No of Cores ……… Construction ……..…… Conductor Type & CSA……..…

‘From’ Equipment …………………….. ‘To’ Equipment …………..…………..…

Inspection Location (room/tray identification point) ………………………………...

Manufacturer ……………………………… Year Installed ……………..…………….

<table>
<tr><td>Operating History</td><td>Comments</td></tr>
<tr><td>Maximum Cable Rating (Amps)

Design Current (amps)

Recorded Loading (amps)

Operational Duty Cycle (%)</td><td></td></tr>
<tr><td>Cable Condition</td><td></td></tr>
<tr><td>Physical Appearance of sheath

Physical Appearance of Cable Core Insulation: ‘From’ End

‘To’ End

Ambient Temperature

Sheath Temperature

Physical Condition of Supporting Steelwork</td><td></td></tr>
<tr><td colspan="2">General Observations and Comments:</td></tr>
</table>

Cable inspection guidance check sheet

3.96. The following lists contain inspections and measurements that should be conducted on power cables to ascertain the condition and serviceability of cable assets that have a typical service life of 40 years. It is recommended that the data obtained from such inspections is stored in a database to enable progressive examination of the condition assessment data.

Paper insulated lead sheathed cables up to 11kV

3.97. Disturbing or moving old paper insulated, lead sheathed, cables will often lead to failure of the cable. This is due to voids being created between the lead sheath and the cable cores which may not be refilled with compound. The general advice is to avoid disturbing old cables.

3.98. Walk down inspections:

a) All cables should be examined for evidence of mechanical damage or crushing.

b) Where cables are installed above ground all cable supports and cleats should be examined for corrosion, security and to check for fretting between the cable and the supports. Particularly check for cracking on the surface of the lead sheath in the region of the support position.

c) On older cables with armour wires the outer serving may have degraded, if so any exposed armour wires should be examined for corrosion.

d) Cable terminations should be examined for evidence of corrosion or overheating at the bolted connections, where accessible.

e) Compound filled termination boxes should be examined for corrosion and compound leakage.

f) If accessible, the lead plum at the entry to termination boxes should be examined for signs of cracking or compound leakage.

g) Bonding leads and earth straps should be examined for corrosion, mechanical damage or overheating.

3.99. During routine maintenance:

h) In air terminations should be examined for evidence of tracking, surface cracking or crazing and partial discharge (PD) activity. White powder in the crutch area of heat or cold shrink terminations is a sign of discharge activity occurring.

i) Heat or cold shrink type terminations within enclosures should be examined for evidence of discharge activity and tracking. Corrosion within air filled termination boxes can be an indication of discharge activity. Ozone generated by the discharges can combine with atmospheric moisture to form nitric acid.

j) PD monitoring equipment could be installed on selected circuits where walk down inspections have identified features. Suitable equipment can be used to monitor discharge activity during normal operation.

k) Continuity and resistance measurements on bonding leads and earth strap connections can be used to monitor the condition of these connections.

3.100. Additional condition assessment inspections:

l) Paper insulated cables should be partial discharge mapped during an annual routine outage. This will identify regions of high discharge activity within a cable circuit and enable progressive monitoring with time. The discharge test results can be used to prioritise circuits for a renewal programme. It should be noted that HV DC withstand tests are not recommended on old cables that are still in service.

m) PD detection using a hot-stick or ultrasonic probe should be carried out on terminations that are not contained within compound filled boxes. For in air terminations, PD activity levels will be affected by the ambient conditions (humidity). The discharge levels recorded can be used to prioritise circuits for a renewal programme.

<u>PE or PVC sheathed EPR & XLPE insulated power cables up to 11kV</u>

3.101. The weakest components in EPR and XLPE cable circuits are the cable joints and terminations. These items are susceptible to moisture, thermo-mechanical movement and external mechanical damage. The general advice is not to move a cable near to a joint or termination position. Heat shrink terminations should be reheated (reshrunk) following any disconnection procedure.

3.102. Walk down inspections:

a) All cables should be examined for evidence of mechanical damage, crushing or movement of armour wires.

b) Where cables are installed above ground all cable supports and cleats should be examined for corrosion, security and to check for fretting between the cable and the supports.

c) PVC / PE cable sheaths should be inspected for crazing, cracking, etc.

d) Cable terminations should be examined for evidence of corrosion or overheating at the bolted connections, where accessible.

e) Bonding leads and earth straps should be examined for corrosion, mechanical damage or overheating.

3.103. During routine maintenance:

f) In air terminations should be examined for evidence of tracking, surface cracking or crazing and PD activity. White powder in the crutch area of heat or cold shrink terminations is a sign of discharge activity occurring.

g) Heat or cold shrink type terminations within enclosures should be examined for evidence of discharge activity and tracking. Corrosion within air filled termination boxes can be an indication of discharge activity. Ozone generated by the discharges can combine with atmospheric moisture to form nitric acid.

h) Continuity and resistance measurements should be conducted on bonding leads and earth strap connections. The measurements can be used to monitor the condition of these connections over time.

3.104. Additional condition assessment inspections:

i) PD detection using a hot-stick or ultrasonic probe should be carried out on terminations that are not contained within compound filled boxes. For in air terminations, PD activity levels will be affected by the ambient conditions (humidity). The discharge levels recorded can be used to prioritise circuits for a renewal programme.

j) Cables with XLPE insulation should have very low levels of partial discharge activity that are likely to be below the background noise level. PD mapping may identify discharging joints or terminations. Tan δ measurements are suggested as a means of monitoring the bulk condition of the insulation. Comparisons can be made between results from similar cables to indicate whether any cable has a significantly higher Tan δ than others.

PVC, XLPE, paper insulated LV cables < 1kV

3.105. Walk down inspections:

a) All cables should be examined for evidence of mechanical damage, crushing or movement of armour wires.

b) Where cables are installed above ground all cable supports and cleats should be examined for corrosion, security and to check for fretting between the cable and the supports.

c) PVC / PE cable sheaths should be inspected for crazing, cracking, etc..

d) Cable terminations should be examined for evidence of corrosion or overheating at the bolted connections, where accessible.

e) Bonding leads and earth straps should be examined for corrosion, mechanical damage or overheating.

3.106. During routine maintenance:

f) The cable insulation resistance (IR) should be measured. Armoured cables should be measured phase to earth. Unarmoured cables should be measured between phases. Cables with a metallic earth return and a PE or PVC over sheath should also be measured between the armour wires and earth.

NB: IR measurements can vary greatly with temperature, e.g. the resistivity of PVC drops by approximately 50% with a 10° increase in temperature. It is recommended that repeat measurements are conducted at a similar time of day to minimise variations is environmental conditions, e.g. the first measurement of the day at 08.30. Ambient conditions at the time of measurement should be recorded to enable temperature normalisation of the results.

g) Cable terminations that are shrouded during normal operation should be inspected for signs of corrosion and overheating.

h) Continuity and resistance measurements should be conducted on bonding leads and earth strap connections. The measurements can be used to monitor the condition of these connections over time.

3.107. Additional condition assessment inspections:

i) Where electrical testing has proved inconclusive, non-destructive mechanical or micro-sample chemical testing should be considered. Research on assessment techniques for LV cables have been on-going for the last 20 years. There is only one commercially available non-destructive condition monitoring test method that can be performed on LV cable systems, the Mechanical Compressive Modulus (cable Indenter). The Indenter monitors the mechanical property changes of the insulation/jacket material. The indenter can be used as an ongoing trending tool, or if available the degree of ageing can be determined by comparing the compressive modulus measured on the installed cables to the results from cables that have been artificially aged.

j) If the above testing is not appropriate or available, or if a cable is experiencing an increased number of faults or routine inspections are indicating significant changes in parameters, then it is recommended that samples of the cable are sent to a cables specialist for residual life assessment. Where appropriate the study should include a visual assessment of the cable build, a mechanical assessment of the insulation (bend test, hot set measurement, thermo oxidative tensile test), electrical assessment (accelerated life test), chemical analysis of the insulation (including oxidation induction time analysis (OIT) measurement to determine whether the insulation stabilisation package has been exhausted) and a review of the cable / maintenance service history.

Annual review of cable assets (I&C, LV and HV)

3.108. It is recommended that the cable inspection schedule is formally reviewed annually to determine whether there is an increasing number of faults occurring on a particular route or particular type / age of cable. The review enables the continued development of cable maintenance and replacement schedules, identifying trends and progressive deterioration and any unexpected degradation.

Cable faults

3.109. Cable faults should be investigated at the time of occurrence. The nature of the fault mechanism should be documented within the cable inspection schedule for future reference.

Frequency of inspections

3.110. Table 3.5 provides the recommended inspection frequency for various cable types.

Cable Insulation Type	Inspection Period (routine)
PVC Power cable	6yr
PVC Multicore or Multipair	
Control Cable	6yr
Mineral Insulated	6yr
PE (polyethylene), XLPE (cross linked PE) and EPR (ethylene propylene rubber)	6yr
Paper Insulated	10yr
Trenched cables	6yr
Special Cables	Typically 6yr Depends on type and environment

Table 3.5 Guidance on inspection frequencies.

3.111. These inspection frequencies are meant as an initial guide to the maximum inspection interval for each cable type/environment. The inspection frequencies should be adjusted accordingly based on the condition of the cables or the environment they are installed in.

Acceptance criteria

3.112. The acceptance criteria for the inspection activities are covered in Section 3.1.2.1.

<u>Additional monitoring and testing provisions provided to PWRs</u>

3.113. To provide an additional means of understanding any ageing mechanisms and predicting remaining life, samples of each type of cable are retained on site. These are held in two equipment storage facilities. One is located inside containment and subject to the same environment to other cables subject to radiation exposure. A second facility containing the same sample types is contained in a similar facility located on site in the stores area and so free from radiation exposure. Small pieces can be cut from these samples during the stations life to assess or help identify possible ageing mechanisms not revealed in other installation inspections or be used to help explain any deterioration discovered during installation inspections.

3.114. With the objective of complying with our NSAPs an environmental monitoring system has been installed at Sizewell B to gather environmental data during the life of the plant. This will provide valuable information in understanding the actual conditions experienced by the cabling installation. It can then be used to help in the validation of qualified life calculations and any life-time extension programs.

3.1.3.2 Hinkley Point C (NNB)

3.115. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

3.116. In the UK an inspection and monitoring strategy has evolved which sets out how cabling installations inside Nuclear Power Stations will be addressed. This identifies walkdown inspections, periodic intervals and particular signs to look out for. This strategy will be adapted and revised as appropriate for use on EPR™ installations in the UK. In addition key preventive and remedial actions will be considered.

3.117. In addition to visual inspections additional means of understanding any ageing mechanisms and predicting remaining life will be implemented. For example: It is currently being considered that samples of each type of cable could be retained on Site in two dedicated equipment storage facilities. One located inside the radiological containment building housing the reactor and thus exposed to the same environment as service cables. And a second dedicated facility containing the same sample types will be contained in a similar facility located on Site and so free from radiation exposure. Small pieces can be taken from these samples during the stations life cycle to assess or help identify possible ageing mechanisms not revealed in other installation inspections or be used to help explain any deterioration discovered during installation inspections.

3.118. An environmental monitoring system is being considered as a possible method to gather environmental data during the life of the plant. This will provide valuable information in understanding the actual conditions experienced by the cabling installation. It can then be used to help in the validation of qualified life calculations and any life-time extension programs. This monitoring system will also be used to gain information on both the normal installed cable installation and those cables that are contained within a cable tray wrapping system. Cable tray wrapping is installed on the EPR™ to provide functional isolation, from a fire perspective, of diverse lines of protection and, where necessary, to reduce fire load.

3.1.4 Preventive and remedial actions for electrical cables

3.1.4.1 Operating reactors (EDF-NG)

3.119. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

3.120. The main preventative and remedial actions for electrical cables are identified in Table 3.6 below.

	Cable Condition	Cause	Immediate Action	Long-Term Action
1.	Bubbles in the jacket surface	Moisture under jacket (Manufacturing defect)	Replace cable at next outage to eliminate manufacturing defect if insulation is similar and severely affected.	None
2.	Blisters in jacket surface	Severe external heat from a local source	Identify heat source and either remove it or install a shield between cable and source. Evaluate severity of damage; if underlying insulation is damaged, replace cable.	If cable has not been replaced, inspect at next outage to monitor for any change.
3.	Brittle	Excess heat (current loading or external source) Excess radiation	Identify heat source: If it is ohmic heating, review circuit loads. If the source is external, either remove it or install a shield. Evaluate effect on safety/operability. Replace cable if insulation is cracked or very brittle. If some flexibility remains, continue use. Identify radiation source. Evaluate effect on safety/operability. Replace	Inspect at next outage to monitor for any chang e.

	Cable Condition	Cause	Immediate Action	Long-Term Action
			cable if insulation is cracked or very brittle; otherwise, install shielding between source and cable. If some flexibility remains, continue use.	
4.	Burned	Excess heat from current load or external source (hot spot)	Identify heat source and extent of damage. If burn extends for a considerable length, then current load is excessive, resize and replace cable. If burn area is localised, then eliminate hot spot or install shield between cable and source. If burn is at or near the crimped connection, determine if lug was properly installed.	Inspect at next outage to assess rate of change.
5.	Cracked	Severe ageing Mechanical damage Ultraviolet radiation from fluorescent lights	Determine if crack is only in jacket or if it extends to the insulation. If crack is just in jacket, then apply a temporary repair (tape/heat shrink) to jacket. If crack is in insulation, then replace cable. Identify source of damage.	Inspect at next outage for any change. Replace cable if cracking has extended beyond original area.
6.	Crystals (salt)	Boric acid drip	Locate source and stop. Clean cable.	Inspect at next outage for reoccurrence.
7.	Discoloured	Initial ageing	None if not accompanied by	Inspect at next outage

	Cable Condition	Cause	Immediate Action	Long-Term Action
		Chemical spill	brittleness or cracking. Review data for interaction between material and chemical. Clean cable or replace.	for any change.
8.	Dull	Normal – new Chemical spill	None Review data for interaction between material and chemical. Clean cable or replace.	Inspect at next outage for any change.
9.	Green Ooze	Plasticiser from PVC	Clean contacts where wire was attached. If the ooze was produced from PVC insulation, then remove wire and install new wire. If the ooze was produced only from a PVC jacket, then no cable replacement is required.	None.
10.	Limp	Ageing process for certain materials Chemical spill	None if integrity remains. If softening is severe, replace. Review data for interaction between material and chemical. Replace if necessary.	Inspect at next outage to look for any change.
11.	Oily	Manufacturing defect Oil/Chemical spill	No action required. Identify oil or chemical and review data for interaction with material. Clean cable or replace if cable material is susceptible to oil or chemical contamination.	None.
12.	Red goo at splice	Adhesive from heat-shrink material	No action required. This is normal condition in most cases.	Inspect at next outage to look for

	Cable Condition	Cause	Immediate Action	Long-Term Action
		flowed out of splice.		any change.
13.	Shiny	Normal – new Chemical spill	None. Identify chemical and review data for interaction with material. Clean cable or replace.	Inspect at next outage to look for any change.
14.	Slimy	Wetting and biological growth	Identify source of moisture and eliminate it. Clean cable.	Inspect at next outage to look for recurrence.
15.	Stiff	Normal – new Intermediate ageing process for some materials	None. A previously flexible material will have exceeded ~ 50% of its life at this point.	None. Inspect at next outage for any change.
16.	Swollen	Moisture absorption Chemical spill	Identify source and stop. Evaluate if insulation has been affected significantly; if so, replace cable. Identify chemical and review data for interaction with material. Clean cable or replace if cable material is susceptible to oil or chemical contamination.	Inspect at next outage to look for recurrence.
17.	Sticky or tacky	External – chemical or oil spill Internal – plasticiser from PVC (may or may not be green)	Identify oil or chemical and review data for interaction with material. Clean cable or replace if cable material is susceptible to oil or chemical contamination. Evaluate extent of conductor corrosion and replace cable if required.	None. Inspect at next outage to look for change.

	Cable Condition	Cause	Immediate Action	Long-Term Action
18.	White dust	Bloom from PVC jacket	The appearance of bloom can indicate that PVC has aged over 50% of its thermal life. Evaluate ageing using Indenter or stiffness relative to new PVC. If in a very high radiation zone (> 30 Mrad [> 0.3 Mgrays] total dose), white dust is HCI. Material will experience high leakage currents if steam environment occurs. Replace cable if this is a safety circuit.	Inspect at next outage to look for any change.

Table 3.6 The main preventative and remedial actions for electrical cables

3.1.4.2 Hinkley Point C (NNB)

3.121. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

3.122. Though many of the visual inspection procedures included in the existing UK strategy may be implemented at HPC, with adjustments where necessary, additional methods will be introduced to cater for those cables that are contained within cable tray wrapping. This is because the visual inspection procedures developed for the main installation may not be appropriate. These are yet to be developed. The cable tray wrapping encloses all cables on a tray along the entire route within a fire compartment therefore the inspection regime that is implemented will require some modification to cater for access difficulties. It is likely in these cases that visual inspection will be limited and there will be more reliance on Environmental Monitoring. Inspection and monitoring will be undertaken on the installation only if it is not envisaged that any cable samples will be stored within a wrapped environment.

3.2 Licensees' experience of the application of AMPs for electrical cables

3.2.1 Operating reactors (EDF-NG)

3.123. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

<u>Cable failures at Torness NPP</u>

3.124. The most significant experience has been associated with the replacement of the 11kV EPR cables at Torness Power Station with equivalent XLPE cables at a cost of approximately £10m. The programme was initiated following a series of random cable failures that was finally attributed to poor quality controls being applied during manufacture. The online partial discharge monitoring equipment was deployed and used to rank those cables that were at the highest risk of failure and hence should be replaced at the earliest opportunity.

Cable failures at Sizewell B NPP

3.125. Of lesser significance several random failures of 3.3kV cables occurred at Sizewell B Power Station in sections that were buried. Forensic stripdowns were carried out by ERA Technology in Leatherhead, Surrey to establish the likely reason. The evidence showed high moisture levels in the bedding layer that was not attributable to sheath damage, joints or terminations and must therefore have originated during the manufacturing process.

Failure of an 11kV cable box at Hartlepool NPP

3.126. Hartlepool Power Station, in September 2016, Unit Transformer 1A 11kV blue phase cable box was distorted and some of the cold pour 'Guroflex' resin, which acts as an electrical insulator, was found to have been ejected from a split weld, indicating a fault and over-pressurisation had occurred.

3.127. National Grid reported no faults or lightning strikes on the local transmission system, nor was there any switching taking place in close vicinity to the station that may have led to an overvoltage transient.

3.128. The root cause of the fault points towards the Guroflex not being applied correctly. In particular there was evidence of poor adhesion around the HV bushing due to the presence of Vaseline, or a similar substance, which must have been mistakenly applied prior to the first fill of the box. It is believed that this would have led almost immediately to the onset of partial discharge activity, which over a prolonged period degraded the insulation to such an extent that a flashover to earth occurred.

3.129. As well as at Hartlepool Guroflex has been used elsewhere in the fleet and therefore the extent of condition is an important consideration.

3.130. As has been stated it is believed the fault occurred as a consequence of poor / inadequate controls in place at the time the cable box was filled which led to partial discharge activity, deterioration of the insulation and eventual breakdown and flashover. As a consequence of this event EDF-NG is accelerating the rollout of the online partial discharge monitoring equipment across the fleet.

3.131. In the interim EDF-NG has taken confidence that the risk of further failure is small based on the testing carried out thus far, no widespread or systematic move from bitumen to Guroflex having occurred across the fleet, and finally due to the lack of similar discharge activity on the two other phases associated with the Hartlepool Unit Transformer 1A.

3.132. Notwithstanding the above failures, the strategic change in the monitoring of cable ageing from one that was carried out as part of routine maintenance and inspection activities or during replacement of defective plant to one that is more systematic has proven to be effective. The development of the Technical Guidance Note has identified best practices and helped raised the standard across the fleet. The evidence of fewer failures and improving ER index (ERI) and unplanned trips across the fleet can in part be attributed to this strategic change.

Neutron Flux Cables

3.133. Neutron flux cables have been replaced at several stations including Heysham 1 and Hinkley Point B.

Recommendation be EDF-NG

I&C Cables

3.134. During this review it was noted that although the existing Cable Condition Monitoring Technical Guidance Note doesn't differentiate between power cables and I&C cables, it was found that the I&C In-Service Management Company Technical Standard doesn't refer out to it; this shortfall will be closed at the next revision, which will be issued by 31 March 2018.

3.2.2 Hinkley Point C (NNB)

3.135. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

3.136. The existing UK fleet undertakes regular inspections as discussed earlier and has experience of random cable failures, some attributed to inadequate quality controls being applied during manufacture. As an example, in such instances when issues were identified, partial discharge monitoring equipment was used to rank those cables that were at the highest risk of failure and hence required replacement. In other cases forensic stripdowns have been carried out by outside specialists to establish the likely reason for failure. All findings are shared across the fleet to ensure a better understanding of ageing issues.

3.137. This will be complemented by OPEX from the EDF French fleet both that currently in operation and FA3 when it comes on line.

3.138. Actual ageing will be compared to predicted ageing and explanations for apparent differences sought.

3.139. In conclusion these are very early days in the establishment of an AMP for cabling in the UK EPR^{TM}. However there is a high degree of awareness of the current issues and procedures via the existing UK fleet. That knowledge will be built on by introducing French fleet experience to initially build up a programme and keep it informed and up to date by incorporating internal and external OPEX and good practice.

3.3 Regulator's assessment and conclusions on ageing management of electrical cables

3.3.1 Criteria and standards for assessing the licensee's ageing management programme of electrical cables

3.140. The main criteria and standards for assessing the licensee's overall AMP are, as described in section 2.7.1, the IAEA Safety Guide NS-G-1.12, the WENRA Safety Reference Levels (SRL) Issue I (Ref 2) and ONR's Safety Assessment Principles (SAPs, Ref 3).

3.141. In addition to the above, the following criteria and standards for assessing the licensee's AMP on electrical cables have been used:

- IAEA IGALL AMP201 Insulation Material for Electrical Cables and Connections Not Subject to Environmental Qualification Requirements (Rev 2015) (Ref 34),
- IAEA IGALL AMP202 Insulation Material for Electrical Cables and Connections Not Subject to Environmental Qualification Requirements Used in Instrumentation Circuits (Rev 2015) (Ref 35),

- IAEA IGALL AMP209 Ongoing Qualification of Electrical and I&C Components Relevant to an Environmental Qualification (Ref 36),
- IAEA IGALL AMP2010 Condition Monitoring of Electrical and I&C Cables Subject to Environmental Qualification Requirements (Ref 37),
- IAEA TECDOC 1188 In-Containment Cables (Ref 38).

3.3.2 ONR's assessment of EDF-NG's ageing management programme for electrical cables

3.142. In undertaking this assessment, ONR considered the implementation of the ageing management programme (AMP) for electrical cables established at EDF-NG's operating reactors sites, where the approach taken is by means of a cable condition monitoring Technical Guidance Note (TGN). The detailed advice provided within the TGN for the purposes of cable condition monitoring is described, by EDF-NG, within section 3.1.

3.143. EDF-NG's approach to the ageing management of electrical cables is integrated into its management system as described and agreed as acceptable in section 2.7.3. It utilises Maintain Design Integrity (MDI) and Equipment Reliability (ER) as the key processes for managing ageing with Technical Governance (TG) as a supporting process. In considering ageing of electrical cables, TG is the key process, which specifies and monitors the engineering standards and guidance for the design and control of the plant. This includes two main types of document:

  - Company Technical Standards (CTS) which are mandatory;
  - Technical Guidance Notes (TGN) which provide company guidance on how to comply with the CTS and are not mandatory

3.144. EDF-NG's key standards and guidance documents relevant to cable ageing management are:

- In-Service Management of I&C Cables CTS (ISM-CTS)
- Cable Condition Monitoring TGN (CCM-TGN)
- Neutron Flux Detection TGN (NFD-TGN)

3.145. The second series of AGR periodic safety reviews circa 2006 identified that the condition of all cable types and voltage levels were not being routinely monitored. They identified that only cables providing power or control functions to those plant items or equipment that were on a maintenance programme were subjected to regular condition monitoring and this did not include all cable types and voltage levels; and inaccessible or buried cables. Furthermore, EDF-NG recognised that the condition monitoring was limited to Insulation Resistance and Polarisation Index tests. Therefore to address this shortfall EDF-NG produced its CCM-TGN and made arrangements for it to be implemented at each of its operating reactor sites in order to enhance its cable ageing management processes.

3.146. ONR routinely inspects the implementation of the cable ageing management processes when undertaking the following regulatory actions:

- System Based Inspections
- Licence Condition Compliance Inspections
- Reactor Maintenance Outages.

3.147. The CCM-TGN sets out EDF-NG's approach for electrical cable condition monitoring at all of its operating reactor sites and standardises the requirements for cable condition monitoring, assessment and, where necessary, ageing degradation remediation. The TGN is used by EDF-NG's electrical engineers, officers or coordinators undertaking activities associated with electrical cable condition monitoring. It is maintained in accordance with EDF-NG's established governance review arrangements. The CCM TGN covers all types of cables in use at EDF-NG's operating reactor sites and covers cables that are both accessible and inaccessible. Electrical cables at EDF-NG's operating reactor sites are divided into five groups for ageing assessment as described in section 3.1.1.1. The groupings are based on insulation type, which is normally the component of cable construction that can be used to determine the onset of ageing degradation.

3.148. The CCM-TGN provides detailed advice on: the selection of cables and cable routes for assessment; an appropriate inspection schedule that identifies those cables and cable routes selected; the form of assessment and the recording and retention of findings. All of these provide a good indication on the condition of all cables and cable routes.

3.149. Advice is provided on the recognition of the symptoms of ageing degradation mechanisms such as heat, radiation and manual manipulation. Advice is also provided on recommended remedial actions and these range from increased monitoring frequencies to cable replacement, for all identified symptoms.

3.150. ONR considers that the CCM-TGN provides a suitable representation of the criteria and standards for monitoring the condition of cables. Also the TGN confirms that EDF-NG's AMP for cables is consistent with the IAEA IGALLs listed in paragraph 3.141.

3.151. ONR has undertaken inspections at a sample of EDF-NG's operating reactor sites to assess the implementation of the CCM-TGN requirements. During these inspections, ONR found that:

- Suitably qualified and experienced personnel routinely undertake periodic cable condition monitoring assessments using the advice provided within the CCM-TGN.
- Cables and cable routes are assessed in accordance with an established inspection schedule that identifies all the cables and cable routes to be inspected in EDF-NG's processes.
- Initial cable condition monitoring assessments were targeted on a basis that exceeded the advice provided within the CCM-TGN. Assessments covered the extent of the operating reactor's cable installations, including all cable types and all cable routes on representative samples of all insulation types.
- The targeting and focus of cable condition monitoring assessments is enhanced at a rate that is commensurate with the occurrence of the effects of symptoms of ageing degradation mechanisms.
- Cable condition assessments are undertaken on a periodic basis that is commensurate with the extent of the effects of the symptoms of the ageing degradation mechanisms identified. Detailed condition assessments are recorded and retained for the purposes of trending the effects of the symptoms of ageing degradation mechanisms and anticipating the implementation of remedial actions.
- Evidence that remedial actions had been implemented for the identified symptoms of ageing degradation mechanisms.

3.152. Based on the findings from these inspections, ONR found: that the periodic cable condition monitoring advice contained within the CCM-TGN was adequately implemented and is satisfied that: assessments are appropriately targeted and undertaken at appropriate periodicities; cable condition assessments are appropriately recorded and retained; remedial actions undertaken are appropriately developed and implemented. ONR is therefore content that CCM-TGN provides EDF-NG with an adequate means of managing cable ageing.

3.153. ONR found that each operating reactor site interprets and implements the advice provided within the CCM-TGN differently. Given this variation in the implementation of the TGN, ONR considers that implementation of a process to review cable condition monitoring practices and outcomes across all operating reactor sites would enable EDF-NG to establish, so far as practicable, consistency in cable condition monitoring and would offer a means of identifying any differences and areas of best practice. ONR has agreed with and EDF-NG proposal that these should be completed by the end of 2018.

3.154. EDF-NG's own assessment of its approach to ageing management of cables identified that a ISM-CTS does not make reference to the CCM-TGN. Without appropriate referencing, the requirements from the ISM-CTS or the advice from the CCM-TGN could be missed during implementation. In identifying this shortfall, EDF-NG has committed to close it by the next periodic revision of the ISM-CTS, which is due by the end of March 2018.

3.155. The ISM-CTS, which sits above the TGN hierarchically, sets out the in-service management arrangements to be applied to I&C cables. The ISM-CTS is intended to be used in conjunction with all other appropriate TGNs referenced within it. The in-service I&C cables management arrangements provided in ISM-CTS describe applicable testing methods, together with a range of acceptable criteria that can be used to indicate I&C cable condition.

3.156. The advice provided in the CCM-TGN considers the applicable ageing assessments based on insulation type and provides details of the symptoms of the effects of ageing degradation mechanisms relating to these cable types.

3.157. ONR found that the advice provided within the CCM-TGN focuses on the symptoms of ageing degradation mechanisms that could manifest on all electrical cables. The advice provided defines the routine visual inspection activities that could identify if symptoms of ageing degradation mechanisms are present in a cable's insulation. ONR found that additional advice is provided on the symptoms of ageing degradation mechanisms of cables associated with neutron flux detection within a NFD-TGN that supports the ISM-CTS. This advice defines the routine in-service tests and the analysis to be undertaken that would also identify if symptoms of ageing degradation mechanisms were present in neutron flux detector installations.

3.158. ONR considers that all of the advice provided within the NFD-TGN that supports the ISM-CTS and the CCM-TGN should be considered when making an assessment on whether symptoms of ageing degradation mechanisms are present in neutron flux detection cable installations.

3.159. ONR considers that the requirements and advice provided on the symptoms of ageing degradation mechanisms within both the ISM-CTS and the CCM-TGN to be appropriate. However, notwithstanding this, ONR recognises that there is a potential to create a shortfall in the implementation of the NFD-TGN. Therefore ONR considers that EDF-NG should review the CCM-TGN and ensure that appropriate in-service I&C cable condition monitoring advice is detailed within it by the end of its next three-yearly document review, which is due in December 2019.

3.160. ONR concurs with EDF-NG's finding and recommendation on I&C cables, that it needs to ensure that the CCM-TGN needs to differentiate between power cables and I&C cables. ONR recognises the commitment to close this shortfall and notes that a key deliverable has been raised on this matter and is due for closure by the end of March 2018. This has been raised as a finding in this report.

3.161. ONR is aware of all of the cable failures experienced by EDF-NG, as listed in section 3.2.1, and as part of normal regulatory business has considered the extent of the company's investigations, remedial actions and solutions to prevent recurrence. ONR concurs with EDF-NG's responses to these cable faults and findings. ONR considers that the cable condition assessments, the recording and trending of the effects of the symptoms of ageing degradation mechanisms and the proactive implementation of remedial actions to prevent recurrence have been appropriate and effective in anticipating further similar cable failures.

3.3.3 ONR's assessment of NNB's ageing management programme for electrical cables

3.162. The Hinkley Point C site is currently in the preliminary stages of construction and as a result, cable specifications and qualification testing have yet to be finalised. NNB's systems for ageing management are developing at a rate that is commensurate with the current lifecycle stage of the plant; this includes processes for electrical and I&C cable ageing.

3.163. NNB recognises that an appropriate AMP which gives due consideration to electrical and I&C cable ageing is needed. It also recognises that it should include activities for: considering ageing assessments; monitoring; testing and sampling activities; and preventative and remedial actions.

3.164. ONR concurs with NNB's finding that an appropriate AMP is needed and notes that NNB's management systems for managing ageing are at a specific lifetime stage in their development. ONR also notes NNB's commitment to review, develop where necessary and implement an AMP when the cable specification and qualification process has been completed.

3.165. NNB recognises that good practice exists within the current arrangements for ageing management of electrical and I&C cables at EDF-NG's existing fleet of operating reactor sites. NNB has therefore committed to review, develop where necessary and implement similar arrangements when the cable specification and qualification process has been completed.

3.166. ONR will continue to monitor NNB's development of an appropriate AMP for electrical and I&C cabling in a structured and timely manner as part of ONR's normal regulatory surveillance of design and construction.

3.3.4 Overall conclusions on the ageing management programme for electrical cables

3.167. For EDF-NG, ONR recognises that the licensee has a good process for managing electrical and I&C cable ageing on its operating reactor sites. EDF-NG provides a ISM-CTS, a CCM-TGN and NFD-TGN for this purpose. ONR agrees that the CCM-TGN is consistent with international standards and criteria and concludes that EDF-NG's contribution to this chapter of the national assessment report demonstrates that EDF-NG has an adequate AMP for cables subject to addressing the findings identified from this ONR assessment. ONR's assessment of the AMP at a sample of EDF-NG's operating reactor sites concludes that it is adequately implemented.

3.168. ONR considers that there are three areas where secondary improvements to ageing management of electrical cabling on the operating reactors would be beneficial. EDF-NG should:

- Review the implementation of the technical guidance note for cable condition monitoring at its operating reactor sites to identify areas of good practice and make any necessary improvements to ensure consistent implementation by 31 December 2018 (paragraph 3.153).
- Review the advice given within the technical guidance note for cable condition monitoring to ensure that appropriate advice is given for Neutron Flux Detection cables and make any necessary improvements by 31 December 2019 (paragraph 3.159).
- Update the existing Cable Conditioning Monitoring Technical Guidance Note to differentiate between power cables and I&C cables by 31 March 2018 (paragraph 3.160).

3.169. To ensure that these improvements are implemented in a timely manner, they have been brought together along with those from other chapters in Chapter 9. Chapter 9 identifies a single area for improvement (AFI) for EDF-NG to undertake a programme of improvements and the individual elements necessary to complete that programme are clearly identified, along with dates for their completion.

3.170. For NNB, ONR recognises that Hinkley Point C site is currently in the preliminary stages of construction and as a result, cable specifications and qualification testing have yet to be finalised. NNB's systems for ageing management are developing at a rate that is commensurate with the current lifecycle stage of the plant; this includes processes for electrical and I&C cable ageing. ONR concurs with NNB's finding that an appropriate AMP is needed and notes that NNB's management systems for managing ageing are at a specific lifetime stage in their development. ONR also notes that NNB's commitment to review, develop where necessary and implement an AMP when the cable specification and qualification process has been completed. No specific improvements have been identified for Hinkley Point C.

3.171. The overall conclusion from the assessment of electrical cabling is that both licensees have adequate AMPs, given the specific stages of their lifetimes, but that EDF-NG needs to make a limited number of secondary beneficial improvements.

4 Concealed pipework

4.1 Description of ageing management programmes for concealed pipework

4.1.1 Scope of ageing management for concealed pipework

4.1.1.1 Operating reactors (EDF-NG)

4.1. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

4.2. Whereas Section 2 of this report describes EDF-NG's general arrangements available for the ageing management of all SSCs, this section considers the ageing management programmes for concealed pipework, a component within numerous systems across the stations. The general ageing management arrangements across all systems affected are discussed in Section 2, and will not be repeated here. This section will instead focus on the specific arrangements for the grouping of safety-related concealed pipes.

4.3. It is evident from Section 2 that the safety case (managed by the MDI process) is fundamental to the ageing management of safety-related SSCs. The construct of the safety case is typically at plant or system level, covering normal operation faults and hazards. Each safety case identifies the safety-related SSCs supporting the safety case, and their ageing management arrangements, including the overall maintenance, testing and inspection schedule.

4.4. ER is used by EDF-NG to manage the day to day plant condition and performance in accordance with the safety case. As noted in Section 2 the focus of ER is at system level, and predominantly upon plant and system components where active changes in plant configuration and performance can be readily measured and managed. ER is applied to all systems containing concealed pipework.

4.5. TG is used by EDF-NG to provide additional support to the safety case and to ER, in specifying how the plant is designed and controlled. TG is based around disciplines, groups of systems, structures or components, which provide an effective means of capturing and disseminating discipline based national and international standards, and imparting knowledge from discipline based Chief Engineers. Concealed pipework is designed and controlled using TG and hence the focus of the commentary in Section 4 is on the TG arrangements for the ageing management of concealed pipework.

Scope of ageing management for concealed pipework

4.6. A number of pipework failures across the EDF-NG fleet, on a variety of systems (e.g. Secondary Shutdown (SSD) system, nitrogen plant and CO_2 plant), have raised fleet-wide concerns in recent years about the condition of safety related pipework. These concerns ultimately led to the development of EDF-NG's Inaccessible Systems Programme (ISP), established to identify areas of remedial work and develop in-service inspection requirements for inaccessible pipework. This also incorporated electrical cabling and civil structures, aiming to move towards an integration of effort during inspections, though these aspects are not discussed further here.

4.7. For clarity, the ISP for pipework covers the following.

- Inaccessible (or difficult to access) systems or structures are defined as systems or structures which are visible but not readily accessible for visual inspection. This includes systems or structures which are in an elevated location (e.g. carried from steelwork), routed under floor slabs or in covered trenches, or systems covered with lagging or other coverings, irrespective of location. Such systems require extensive scaffolding, roped access, removal of floor slabs or similar to allow access for visual inspection. It would also be necessary to remove lagging or covering from pipes for proper surface inspection.

- Concealed systems or structures are those where the route is fully known but the system or structure cannot be fully seen. This includes systems in concrete trenches or ducts which are substantially or fully covered by slabs, or, for instance, road surfacing over slabs, but the pipe is not, by design, actually in contact with the surrounding materials, or the ducts themselves. For the purposes of the fleet-wide inspections, concealed systems are categorised as inaccessible trenched systems i.e. pipework in covered trenches.

- Buried systems are pipework where the route is substantially known and the pipework is in contact with the surrounding material, either the natural ground material or trench backfill material. Buried systems can only be inspected following excavation. Similarly "buried structures" refer to the external surfaces of concrete ducts where the duct is within the surrounding ground and the external surface is only visible after excavation. Pipework encased in concrete is not considered "buried pipework" although the definition applied to a particular system will depend on the specific design and configuration i.e. the proportion of system deemed inaccessible, the extent of cast-in puddle pipes etc.

4.8. It should be noted that EDF-NG inspections for compliance with the licence condition on EIMT were already in-place as part of maintenance routines for safety-related systems, whereas non-safety critical pipework was not routinely inspected. The need was therefore identified for a new programme of work to specifically address the integrity of pipework systems that were not normally accessible (i.e. buried, lagged, coated etc.) or not inspected as part of an ongoing process; the new programme , entitled the Corrosion Management Programme (CMP), identifies and executes high priority remedial work, with lower priority managed by normal business and establish in-service inspection requirements. The CMP supplements the existing well-developed inspection programmes for systems such as high temperature pipework. Further discussion on the inspections for compliance with the licence condition is given below.

4.9. In support of this programme, a number of TG guidance notes have been developed, with details given below. These guidance notes align with two main company standards on 'Corrosion Management' and 'Pressure Systems', and a further company guidance note for corrosion risk management. The areas of guidance developed to support the programme are:

- **Guidance for determining and assessing the risk from external degradation mechanisms of pipework systems**. This covers the generic requirements with regards to initiating and implementing a Risk Based Inspection (RBI) strategy for inaccessible pipework systems. It should be noted that inspection recommendations for certain specific systems are given in system-specific guidance. This generic guidance is intended to help Stations develop an effective inspection programme for all pipework systems where no other programmes of inspection can be readily identified. Such pipework systems include those which are difficult to access, for example systems where the pipework is contained within concrete trenches, systems which are insulated, entrenched or encased in concrete. All metallic pipework systems are included in this guidance, as are the pipework supports to these systems. However tanks, vessels, valves, pumps and other in-line equipment and pipework materials are considered elsewhere. Buried pipework is also covered separately in the following guidance.

- **Requirements for buried pipework, data gathering and risk ranking**. This guidance presents the data gathering and risk ranking process for buried metallic pipework, focussing on degradation occurring within the external surface or outer layers of the pipe as this mechanism is deemed the most dominant root cause of pressure boundary failure of buried systems. Internal surface degradation is considered as a secondary mechanism with recommendations provided where appropriate. This document provides guidance and recommendations for a practical approach to data gathering and risk ranking information, with the output then used in the following guidance detailed below to enable the aspects for protection, monitoring and inspection of buried pipework to be managed.
- **Guidance for the protection, monitoring and inspection of buried pipework.** As noted above, this guidance presents the protection, monitoring and inspection requirements for buried metallic pipework. All fluid systems containing buried metallic pipework are covered by this guidance, noting differing plant configurations will result in a site-specific scope of work.

4.10. The above documents do not apply to:

- Non-metallic pipes (concrete, High Density Polyethylene (HDPE), Polytetrafluoroethylene (PTFE) & vitrified clay etc.).
- Other types of metallic and non-metallic buried structures (vessels, tanks, conduits and intake structures).
- Pipework located above ground level, or contained within below ground level concrete trenches.
- Maintenance requirements with respect to maintaining the reliability or functionality of these systems as these requirements are met by statutory and routine maintenance.

4.11. Separate TG documents cover a number of these areas, with the potential for production of documents to cover other areas should the need be identified. Should there be a need to install, for example, HDPE to replace an original system, guidance available at the time would be utilised (and likely specified within the associated safety case). This would include testing requirements as part of construction, in-service inspections, etc.

4.12. A summary of the RBI approach taken across each of the guidance documents is given below. It is worth noting that there are plans in-place for the amalgamation of a number of guidance documents to ensure a consistency in the risk-ranking approach to be followed.

4.13. The steel and cast iron pipes partially encased in concrete, commonly called "puddle pipes", are widely used in the seawater cooling systems across the EDF-NG fleet. In a number of areas, these seawater cooling systems serve essential systems such as Reactor Auxiliary Cooling Water (RACW) and Pressure Vessel Cooling Water (PVCW). The puddle pipes serve as a transition between the above ground pipework and the concrete culverts or alternatively connect pipework on each side of a building wall. At some stations these have been repaired/replaced by using a carbon steel base plate and a polyurethane seal, with the seal forming part of the pressure boundary. This has either been in response to a degraded state or to address longer-term concerns over system integrity. In addition, at some Stations, special make up spools, called template spools, are installed between the puddle pipe and the metal pipework to take up the misalignment between the baseplate puddle pipe and the above ground pipework. Specific guidelines for the inspection of the seawater cooling systems is provided in a separate guidance document for the inspection and testing of puddle pipes and the associated Liebig anchors.

4.14. In addition to the guidance aimed at providing detailed information of risk-ranking, inspections, maintenance etc., there are also routines dictated as part of compliance with the licence condition or environmental legislation for EIMT across the fleet. These are aimed at providing confidence in the particular safety system to satisfy whatever safety case claims have been made against it, in addition to ensuring that any associated environmental risk is appropriately managed (e.g. from degradation of fuel oil pipework). To some extent, these systems sit outwith the RBI approach, with the maintenance schedules mandating the level (and frequency) of inspection. Undertaking the RBI approach for these systems may help to confirm the level of inspection required. However, any associated amendments to the maintenance schedules would need to be appropriately justified within the safety case, utilising appropriate arguments and evidence.

4.15. Focussing on the identified NAR examples, details of the activities undertaken at Hunterston B (as an example) are given in Table 4.1 below. Hunterston B is seen as an appropriate site to focus on as it is one of the older AGRs within the EDF-NG fleet, and generally representative of the other stations. It is recognised that the types of issues seen across the AGR fleet may be different to those at Sizewell B (the only PWR within the fleet). However, consideration of the potential differences and similarities between an AGR and a PWR is given, both in response to adverse findings across the fleet and when developing guidance documents. Focussing on Table 4.1 below, any adverse findings arising from these inspections would be rectified within appropriate timescales, recognising the implications of leaks from these systems (and other similar systems across the fleet). Specific operating experience from these systems (at HNB) is given in Section 4.1.2.1 below.

Table 4.1 Hunterston B examples of MS inspection activities.

System	Inspection and Frequency
Radioactive Effluent (Active Effluent Treatment Plant)	Pipework inspection undertaken on an annual basis (dictated within maintenance schedules).
Essential Diesel Generator Fuel Oil	Pipework inspection undertaken (generally, noting the number of diesels installed at station) on an annual basis (dictated within EMITS).
Essential Service Water (Reactor Cooling Water (RCW))	Walk down inspection undertaken on a three monthly basis, with a more detailed inspection every six years (dictated within MITS).

4.1.1.2 Hinkley Point C (NNB)

4.16. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

4.17. The design of the UK EPR^{TM} aims to minimise so far as is reasonably practicable the use of concealed or embedded piping; the scope of the concealed piping AMP is therefore reduced inherently by this minimisation of concealed piping.

4.18. As per the Technical Specification, this section is structured in such a way so as to identify the scope of ageing management related to concealed pipework in the following aspects:

- Methods and criteria used for selecting concealed pipework within the scope of the ageing management;
- Processes/procedures for the identification of ageing mechanisms related to concealed pipework;
- Grouping criteria for ageing management purposes.

Methods and Criteria used for selecting concealed pipework within the scope of ageing management

4.19. No specific methods or criteria used for selecting concealed pipework within the scope of ageing management have been applied by NNB. Rather, AM is applied to all pipework and connections. Three mechanical requirement levels (M1, M2 and M3) are defined for pressure retaining components with the demonstration of an appropriate level of mechanical requirements achieved through application of a combination of safety class and barrier role:

- M1 requirements:
    - The component forms the Reactor Coolant Pressure,
    - The component is a High Integrity Component
- M2 requirements:
    - The component performs a barrier role. The component is required to maintain its pressure boundary integrity during conditions where the component is not isolated from the primary coolant circuit under fault conditions where cladding damage may have occurred
    - The component forms part of a Reactor Building penetration, unless already identified as High Integrity Component (M1).
- M3 requirements:
    - The component performs a barrier role: its failure could potentially, under normal conditions, fault or hazards conditions, lead to a discharge of radioactivity significantly greater than that existing in the surrounding environment

4.20. Where M1, M2, M3 do not apply, “NR” (No Requirements) is used to show that the above rules were considered but do not apply. To ensure the functional capability of mechanical components, appropriate codes or standards are applied in the design and manufacturing of the equipment so that the component quality is appropriate to the function that it provides. The mechanical requirements include provisions for the application of a QA programme, qualification required for specific operating conditions, seismic qualification, and periodic testing, in-service inspection. The mechanical requirements M1, M2 and M3 relate directly to the level of design code or standard to be applied. For some cases, supplementary requirements lead to upgrade the design code or standard to be applied compared to the mechanical requirements. A design quality requirement Q is introduced, which relates to the level of design code or standard to be applied, with the following fundamental rule: Q≥M. The mechanical requirements for pressurised equipment imply the following design codes / standards:

- M1 requires the application of RCC-M1;
- M2 requires the application of RCC-M2 or ASME III with supplements or German Nuclear Safety Standards Commission (KTA) with supplements; and
- M3 requires the application of RCC-M3 (or another nuclear code), or European Harmonised Standards with supplements or any code compliant with Pressure Equipment Directive 97/23/EC, with supplements.

4.21. The application of RCC-M code or equivalent ensures there are sufficient design margins to provide adequate integrity and mechanical stability for the components. A limited number of safety classified components will not be designed according to RCC-M (or equivalent), but similar high standards will be adopted. Typically, in such cases, a well-established design is available and a change of design code would be counter-productive. For some cases, supplementary requirements lead to upgrade the design code or standard to be applied compared to the mechanical requirements. The application of the above codes and standards leads to the implementation of ageing management as part of the design of the UK EPR™. For example, the RCC-M contains the following provision for RCC-M1, RCC-M2 and RCC-M3:

> *"When corrosion or erosion is expected, the wall thickness of the piping shall be increased over that required by other design requirements. This allowance shall be consistent with the service life specified for the piping."*

4.22. Pressure Equipment Directive 97/23/EC contains the following requirement:

> *"The pressure equipment must be designed for loadings appropriate to its intended use and other reasonably foreseeable operating conditions. In particular, the following factors must be taken into account: …corrosion and erosion, fatigue, etc…."*

4.23. The systematic application of safety classification and categorisation principles ensures that pipework and connections have sufficient design margins to provide adequate integrity and mechanical stability for their service life.

4.24. The scope of ageing management of concealed pipework is inherently minimised by the design of the UK EPR™ which aims to minimise as much as is reasonably practicable the use of concealed or embedded piping. The need to take into account the specific degradation risks posed to concealed pipework during the design stage is therefore reduced. This design philosophy is demonstrated by the construction of the Technical Galleries. The Technical Galleries, a series of large buried concrete structures which house and distribute all vital mechanical, electrical and process services across the site, ensure that services critical to the successful operation of the UK EPR™ are readily accessible and visible. The design of the Technical Galleries has been developed to ensure that there is sufficient space to accommodate the services contained within the Gallery structure, including ensuring there is sufficient space for routine access to the Gallery and sufficient space to enable maintenance of the systems and components contained within. The use of Technical Galleries for the majority of the pipework means that the integrity of the pipework and the Galleries themselves can be inspected on a routine basis. The leak tightness of the Technical Galleries is assured via the following means:

- Limitation of stress in the steel reinforcement to control the degree of leakage,
- Use of waterproof movement joints, regularly spaced to minimise cracking derived from shrinkage.
- An internal drainage system created by laying the gallery floor with both transverse and longitudinal falls.
- Use of detailing that does not concentrate cracking in the concrete structures.

4.25. With regards to those Galleries that have a secondary containment role with respect to radioactive liquids, full reliance against the release of hazardous substances is placed on lining the internal drainage system with a protective (decontaminable) paint coating. Waterproof movement joints comprising of waterbars and sealants will be installed between Gallery sectional units and at the interface with adjacent building structures. The function of the sealant (caulking) is to limit both water ingress (i.e. external flooding entering the gallery) and water egress (i.e. internal flooding leaving the gallery and entering the groundwater). The caulking is located on the inner side of the gallery, which is accessible and thus enables the caulking to be inspected, maintained and repaired/replaced. The caulking is therefore considered to provide the primary barrier against leakage (both ingress and egress) through the movement joints. Piping and connections that are routed through the Technical Galleries are not considered to be concealed and are excluded from this NAR.

4.26. Layout Rules embedded into the design process of the UK EPR^{TM} ensure that requirements concerning accessibility, maintainability and the ability to inspect systems and components are systematically taken into account during the installation phase. Some examples of how these Layout Rules contribute to the minimisation of the eventual inventory of inaccessible piping at the UK EPR^{TM} are given below:

- The pipes that are subject to a high level of monitoring or tests at high frequencies must be as accessible as possible.
- The movement area around an active pipe that has to undergo in-service inspections must be designed to allow access for the equipment inspection.
- The pipes that form part of a layer and are subject to in-service inspections must always be located at the edge of the layer, preferably in the lower part (this ensures that the inspection operations can be performed from a floor or a platform)
- Insulated pipes are positioned so that they remain visible and can be monitored along the whole length of their route.
- The pipes must be routed in such a way as to satisfy all the conditions for: commissioning, operation, erection and dismounting and potential maintenance and inspections

4.27. These Layout Rules also contain penetration requirements and stipulate minimum distances between sleeves and the pipework being routed through the sleeve so as to ensure adequate access for subsequent inspection.

4.28. Ageing management will be applied during the operational phase of the UK EPR^{TM} by maintenance and surveillance programmes and examination, maintenance and inspection routines based on the safety classification component lists. Adequate arrangements will be implemented for the inspection and maintenance of pipework and connections not readily accessible for visual inspection (such as that under insulation). Through-life inspection, examination, testing and maintenance will be required for safety critical pipework during the operational life of the UK EPR™ as per Licence Condition arrangements. Full details of the requirements relating to Examination, Inspection, Maintenance, and Testing (EIMT) of the structures will be given in the EIMT programme, which will be finalised at the end of the design process. The appropriate inspection, examination, testing and maintenance arrangements for non safety-critical pipework will be finalised at the end of the design process.

Processes/procedures for the identification of ageing mechanisms related to concealed pipework

4.29. Assurance that ageing mechanisms are correctly identified within the design process is provided for by the considerable nuclear engineering experience held by the Responsible Designer of the UK EPR™, EDF SA.

4.30. The existing UK fleet has considerable experience in the application of AMPs for concealed pipework. Findings are shared across the fleet to ensure a better understanding of ageing issues. Participation of NNB in EDF-NG's 'Corrosion Management User Group' allows for exposure of NNB to corrosion and ageing management issues faced by the existing fleet.

Grouping Criteria for Ageing Management

4.31. No specific grouping criteria are employed by NNB for the ageing management of concealed piping.

4.32. Concealed pipework and connections have been considered for the following NAR examples:

- Containing radioactive effluents;
- Transfer of fuel for emergency power generation;
- Essential service water providing cooling to plant and equipment important to safety;
- Other pipework and connections important for safety.

4.1.2 Ageing assessment of concealed pipework

4.1.2.1 Operating reactors (EDF-NG)

4.33. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

4.34. The general concept of RBI methodology is to determine the likelihood and consequences of component failure and then determine risk significance by combining these two measures. The risk-based inspection programme is then developed by targeting inspection effort on the high-risk items.

4.35. Note. A RBI programme normally identifies inspection requirements in addition to, rather than instead of, those prescribed by existing statutory and safety case requirements. In principle, RBI methodology may be applied to any structure, system or component whose failure would have consequences for plant safety and availability; however in practice the RBI methodology as described is applied almost exclusively to pipework.

4.36. The following steps are identified in any RBI programme, which considers pipework (the aim here is to summarise the methodology used, with subsequent sections attempting to highlight the output and findings from this approach).

- **Determine the scope and level of the evaluation.** The pipework systems and degradation mechanisms to be considered, and the specific RBI methodology to be applied are identified.
- **Sub-system definition.** Where risk of pipework failure is expected to vary within a pipework system, the system is divided into a number of sub-systems where all conditions relevant to the risk assessment are similar. Where sub-systems are utilised in the RBI process, it is important that terminal points associated with each sub-system are identified. The Likelihood of Failure (LoF) and Consequences of Failure (CoF) should then be evaluated for each sub-system. Failure may be defined as an event involving leakage, rupture or a condition that would disable the ability of a pipework system to perform its intended safety or operational function. For pipework, failure usually involves a leak or a rupture, resulting in a reduction or loss of its pressure-retaining capability.
- **Evaluation of the Likelihood of Failure of the piping in each system or sub-system.** Evaluation of LoF provides a measure of the frequency with which a specified failure event would be expected to occur. It is based on a consideration of specific degradation mechanisms and should consider future degradation rates from all these potential mechanisms. The rate of degradation may increase with time as a result of interaction between mechanisms (e.g. corrosion and fatigue). Factors such as overload, misuse, or accidental damage that cannot be easily predicted should be taken into account based on operating experience.
- **Evaluation of the Consequences of Failure of the piping in each system or sub-system.** Evaluating the CoF of pipework systems considers the effects of pipework failure. The analysis of the CoF should focus on the capacity of the failure and subsequent events to cause death, injury or damage to the health of employees and the general population. The consequences of pipework failure to cause harm to the environment and the business, and incorporation of measures to include these risks in the integrity management strategy, may also be considered.
- **Categorisation of the risk significance of all systems or sub-systems.** The purpose of the risk analysis is to identify the potential degradation mechanisms and threats to the integrity of the pipework systems or sub-systems, and to assess the consequences and likelihood of failure. Categorisation of risk is based on consideration of both LoF and CoF.
- **Selection of locations for inspection.** Categorisation of items in terms of risk is used to prioritise inspection requirements. The prioritised inspection plan can then be used to target locations designated as high risk before fitness-for-service could be threatened. Locations designated as low risk may be identified and may justifiably be excluded from further investigation.

4.37. The risk-ranking methodology specified within the guidance utilises best-practice advice given in both the Health and Safety Executive (HSE) in CRR 363/2001 (Ref 39) and European Network for Inspection Qualification (ENIQ) in EUR 21581 EN (Ref 40) documents on risk-based/risk-informed inspections.

4.38. While the methodology detailed above is considered generic, a different approach was taken for buried pipework, with EPRI software (BPWORKS) used to risk-rank the affected pipework systems. This software determines both the likelihood of a leak or break and the consequences of that leak or break and then uses a risk matrix to help to identify the high risk pipework. BPWORKS uses a process called

Dynamic Segmentation, which iterates the data system to continually refine and pinpoint the greatest risk areas along a specific pipe segment.

4.39. BPWORKS is intended to serve as a basis for estimating the probability of corrosion of metallic pipelines, whose external surfaces are in contact with soil or surface water. This utilises plant data, including pipe properties, duty, soil conditions, consequences of failure etc. The probability of corrosion of these items is governed not only by the properties of the materials and the corrosive agents, but also by their design, their size and by external electrochemical effects. Since these parameters cannot always be described with adequate accuracy, the likely corrosion behaviour can only be estimated. Such estimates, in addition to providing information on the type and extent of corrosion to be expected, serve as a basis for deciding which protective measures may or must be taken.

4.40. Following the completion of the risk-ranking exercise, regardless of the technique used, a programme of walk downs, inspections, monitoring etc. then needs to be implemented.

4.41. Recognising the NAR examples from Hunterston B above, the following significant operating experience has been identified in relation to these three systems, presented in Table 4.2. No relevant Hunterston B OPEX could be found for the Active Effluent Treatment Plant so details have been added from a Torness event.

4.42. Noting the OPEX identified in Table 4.2 (and similar events across the fleet), it is considered that future inspections (either as part of routine System Engineer (ER) walk downs or specified by other requirements (e.g. MS)) will monitor the overall system condition. This will ensure the adequacy of the repairs implemented (or identify the need for further remedial activities).

Table 4.2 Hunterston B and Torness examples of operating experience on concealed systems

System	Operating Experience
Radioactive Effluent (AETP)	• No significant Operating Experience, with regard to degradation of this system, was identified at Hunterston B. • At Torness, routine borehole sampling identified elevated levels of tritium. The direct cause was due to leaks on the Tritium Effluent Discharge Lines, with the root cause due to inadequate operational and maintenance routines to specifically consider component design, operational life and ongoing checks for minor leaks on the tritium effluent disposal system. • The leaks originated from gaskets within the system, which had not been routinely replaced. • In addition to the repairs undertaken to address the leaks, a review of system maintenance activities was undertaken, leading to various preventive actions being identified e.g. renewal of coatings, proactive gasket replacement etc.
Essential Diesel Generator Fuel Oil	• Corrosion of the fuel oil transfer pipework from the Bulk Storage Tanks and the Service Tanks was noted on the 415V diesel generators. An Availability Assessment (and associated mitigating actions) was put in-place to enable replacement of the corroded sections of pipework. This was supported by isolation of the Bulk Tanks from the Service Tanks, with the Service Tanks remaining available for the duration of the replacement activities.

System	Operating Experience
	• Routine maintenance identified that the common fuel oil transfer line supporting the 11kV diesel generators was corroded. An Availability Assessment (and associated mitigating actions) was put in place until the affected section of pipework could be replaced.
Essential Service Water (RCW)	• Following a number of issues with the original Cast Iron (CI) RCW pipework, requiring remedial, localised repairs, and work done to replace the aboveground sections of the systems, a project was commencing for the replacement of the buried sections of the RCW. In September 2011, whilst in the preliminary stages of this project, a section of the R3 RCW Buried CI (BCI) main failed. The R3 RCW main was repaired and returned to service, providing the safety case justification for the continued use of the R3 and R4 RCW systems until such time that an accelerated programme of work to replace the RCW BCI pipework was delivered. Subsequently, the BCI pipework was replaced with aboveground pipework using HDPE and glass flake lined carbon steel.

4.1.2.2 Hinkley Point C (NNB)

4.43. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

Concealed Pipework containing radioactive effluents;

4.44. The Nuclear Island contains various pipes embedded in the rafts and floors:

- stainless steel double-wall pipes (and associated inner supports), which are sections of the RPE system (nuclear drains, vents and exhausts) and partly or fully embedded in concrete rafts and floors,
- carbon steel single-wall pipes, which are sections of the SEO system (wastewater network connected to the sewer) and partly or fully embedded in concrete rafts and floors,
- stainless steel single wall pipes, which are sections of the EVR system (containment cooling ventilation system) and partly embedded in concrete wall.

4.45. The RPE equipment consists of two single-wall pipes, one inserted (hereafter called “inner pipe”) in the other (hereafter called “outer pipe”), free one from another. The inner pipes are similar to drains. The function of the outer pipes is to protect the inner pipes against their environment (e.g. concrete), to allow the movements of the inner pipes resulting from loadings they are subjected to (weight, thermal expansion, seismic event, etc), as well as to collect the potential fluid leakages which could come from the inner pipes and transfer them to the appropriate collectors located within sumps. The RPE equipment is designed and manufactured to Q3 class. The SEO equipment consists of carbon steel (P265GH) single-wall pipes, partly or fully embedded in the concrete of building structures. The pipes are similar to drains and are designed and manufactured to Q3 class. The pipes are seamless and are painted on the outside. The EVR equipment consists of stainless steel single wall pipes, partly embedded in the concrete of Reactor Pit wall. The pipes are seamless and designed and manufactured to Q3 class.

4.46. For all of the above, NNB explicitly requires the system designer to take all necessary precautions to protect the equipment from corrosion, paying special attention to galvanic corrosion and the possible impact of erection

4.47. Following appropriate treatment, monitoring and authorisation for discharge, radioactive liquid effluent is transferred to an outfall pond (designated ‘HCA’) through the Liquid Radwaste Monitoring and Discharge System (LRMDS, designated ‘KER’) or Site Liquid Waste Discharge System (designated ‘SEK’) pipes. A small section of this pipework is not contained within the Technical Galleries but instead is routed through a surface trench running along the front of the pumping station to the outfall building, finally dropping down to the main cooling pipes discharge area of the outfall pond. Adequate arrangements will be implemented for the routine inspection of any pipes located within trenches, utilising operational experience feedback of the issues associated with concealed systems, and noting the length of pipe within the trenches will be minimised as far as is reasonably practicable.

4.48. Full details of the requirements relating to Examination, Inspection and Maintenance, and Testing (EIMT) of the structures will be given in the EIMT programme, which will be finalised at the end of the design process.

Concealed Pipework - transfer of fuel for emergency power generation;

4.49. For the UK EPR™, there will be two geographically separated diesel buildings for each unit. Each of the two diesel buildings contains two Emergency Diesel Generator sets (or Main Diesel Generator sets) and one Station Black Out set (or Ultimate Diesel Generator set). The fuel and the pipelines required for the transfer of this fuel to the diesel generator sets are housed within these buildings. As such, the UK EPR™ design contains no concealed pipework for the transfer of fuel for emergency power generation.

Concealed Pipework - Essential service water providing cooling to plant and equipment important to safety

4.50. Steel cylinder concrete pipes (SCCPs) carrying cooling water from the pumps supplying the circuits of the reactor unit and the turbine building are embedded in concrete in the pump house building. SCCPs are used for the CRF (main cooling circuit), SEC (cooling circuit for heat exchangers in the SEC/RRI network), SEN (raw cooling circuit of the SRI) and SRU systems (cold water source of the EVU cooling system).

4.51. The main cooling circuit CRF (condenser cooling) is not safety classified and, as such, is excluded from consideration in this NAR example.

4.52. Preliminary design substantiation work shows that the SEC, SEN and SRU circuits will be designed and manufactured to M3 class. Compliance with a 60 year lifetime of operation is afforded by the addition of a sacrificial 3mm pipe inner steel core thickness.

4.1.3 Monitoring, testing, sampling and inspection activities for the concealed pipework

4.1.3.1 Operating reactors (EDF-NG)

4.53. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

4.54. Section 2 summarises the Ageing Management (AM) approach followed at each of the stations, utilising the ER process. This requires, among other things, ongoing monitoring of system performance, rectification and trending of defects within a system, review and revision of PM routines. As discussed in paragraph 2.88, the output from the ER process is reported to the station Plant Health Committee, aiming to ensure that appropriate actions are taken in response to adverse findings.

4.55. It is recognised that the specific details of the risk-ranking (considering nuclear, industrial, environmental and commercial risk) undertaken may vary from system type to system type i.e. a buried system compared to an inaccessible system. However, the general principles highlighted above are considered generic. In terms of the approach taken for monitoring, inspections, etc., it is considered more appropriate to discuss the different types of pipework in turn.

Inaccessible (or concealed) pipework systems

4.56. A four-stage approach has been adopted for identifying inspection requirements and other measures to minimise the risk of pipework failure due to external degradation. The assessment of risk takes account of both the likelihood and the potential consequences of pipework failure due to external degradation. The four-stage approach below has been adopted. On the completion of each stage of the process, there was a hold point where the scope of work completed and the results and recommendations produced were subject to a Technical Review by a SQEP employee followed by a subsequent review and endorsement or otherwise by a Pipework Assessment Panel. The Pipework Assessment Panel gave due consideration to the results and recommendations, to record either:

- Their agreement to the findings and recommendations.
- Request for more reassessment /inspection information.
- Recommendation that the Station review findings and consider a course of action in accordance with the decision of the Pipework Assessment Panel.

4.57. As noted below, the Pipework Inspection Programme was used for the delivery of these stages, with this programme now closed (albeit with some longer-term recommendations identified). As such, it is considered that the approach detailed below has been followed at each station, with the longer-term action being the delivery of the "ongoing monitoring and inspection requirements".

<u>Stage 1</u>

4.58. Develop a baseline risk assessment for all pipework systems. This involved a desk-based study to compile data and develop a risk-prioritised list of systems requiring further attention. Systems where risk is deemed to be tolerable were excluded from further consideration at this stage.

<u>Stage 2</u>

4.59. Addresses pipework systems where the risk established in Stage 1 is assessed to be intolerable. A refined risk assessment was carried out which is based on more detailed information regarding external degradation, including the findings of a plant walk down. Stage 2 identified pipework systems where risk due to external degradation is tolerable, as well as identifying systems where further action should be taken to minimise risk. The refined risk ranking of Stage 2 identifies inspection requirements for "at-risk" pipework systems were conducted in Stage 3.

<u>Stage 3</u>

4.60. In Stage 3 a more detailed inspection was carried out, which included screening and/or wall thickness checks at identified locations of the "at-risk" pipework identified in Stage 2. The findings of these inspections were then used as input to the fitness-for-service assessments. The results of these assessments were then applied to update the risk model. Pipework systems where the results of assessments can be used to justify a reduction in the assessed risk were excluded from further consideration. For the pipework systems where this is not the case a refined prioritised list of the mitigation factors such as remedial actions and/or ongoing inspection requirements were defined to ensure that risk is minimised.

Stage 4

4.61. In Stage 4, a strategy for the lifetime maintenance of the RBI programme was drawn up and the ongoing monitoring and inspection requirements specified (with future inspection requirements generally ranging from 3-12 years, depending on the output from the RBI, supported by the ongoing system monitoring). Moving forward, these requirements should be reviewed periodically and, if necessary, the Stage 3 RBI programme of work should be repeated taking into account any new data on the degradation of the pipework systems under review.

4.62. The Pipework Inspection Programme was undertaken to deliver this staged approach across the fleet. Details of the closure of this programme are given in Section 4.2 below.

Buried pipework systems

4.63. Utilising the output from BPWORKS and the risk-ranking undertaken, each particular pipework system was identified as High, Medium or Low risk. The High risk systems warrant a proactive approach, with the Medium/Low given a reactive approach. The initial phase of survey is an indirect inspection. This includes above ground surveys (ranging from system walk downs and checks to more targeted NDT) to identify the presence and location of the buried pipework. The survey results, in conjunction with the output of BPWORKS, can identify, in accordance with risk, the specific sites for direct inspection. Direct inspection consists of actual excavation and detailed inspection of selected sections of buried pipework.

4.64. The output and findings from the inspections/excavations are then fed back into the initial risk-ranking exercise, helping to confirm the future inspection requirements.

High risk buried pipework strategy

4.65. Managing the inventory of pipework categorised as High Risk requires a proactive approach in order to maintain safety margins and system availability and integrity. The proactive strategy for all High Risk pipework is split into three phases, with the requirements for each phase discussed below. It should be noted that Phases 2 and 3 are common to both High Risk and Medium/Low Risk pipework.

4.66. Phase 1 – High Risk buried pipework management plans: A management plan was developed and owned by the appropriate system owner, or delegate, for all High Risk buried pipework. It was recognised that a proactive approach may require an increase in the allocation of station resources in the short term. However, the long term advantage of a proactive approach throughout the life of the station justifies this approach both on safety and economic grounds.

4.67. Phase 2 – Investigatory works: The purpose of Phase 2 is to provide evidential assurance that degradation is present prior to the definition of a plan of action. Investigation techniques can be deployed to help determine whether degradation of buried pipework is present and with some techniques its location along the pipe. It may be necessary to review the results of the techniques deployed specifically considering the confidence of the result determined. It is recognised that a range of techniques are available to aid the process of degradation and leak searching within buried pipework, with positives and negatives from each (as discussed later, there are plans being developed to refine the number of techniques available). The TG guidance provides details of a number of “indirect” and “direct” inspection techniques (paragraph 4.92).

4.68. If degradation and its location can be confirmed from the results of this investigatory work, Phase 3 would then be followed.

4.69. Phase 3 – Resolution of issue: Following confirmation that degradation is present at a specific location, a plan of action must be formed to resolve the issue. The solution deployed is to be defined on a case-by-case basis due to the nature of each buried pipe section and the degradation discovered. Solutions for the resolution of buried pipework degradation include the approaches shown below. A benefit assessment should be completed for each option stated. It should be noted that there have been limited examples, if any, of Phase 3 needing to be implemented.

- Excavating the soil around the pipework to enable visual examination and NDT to be carried out on the outside surface of the pipe.
- Breaking into the bore of the pipe in order to carry out visual examination and NDT from the inside surface of the pipe.
- Formulating a fitness for service case based upon the routine and supplementary investigational results.
- Carrying out refurbishment or repair to the pipework.

Low and Medium Risk buried pipework strategy

4.70. Low and Medium Risk pipework does not require a specific management plan (Phase 1) however routine monitoring should be undertaken. This includes periodic walk down, checks on any make-up tank levels, system pressures etc. Where these routine monitoring procedures identify that a buried pipework section may have degraded then Phase 2 above is followed. This differentiates the proactive (High Risk) from the reactive (low/medium risk). It should also be noted that routine monitoring is undertaken on High Risk pipework, to an increased frequency to the low and medium pipework.

4.71. Where degradation is not identified following routine monitoring, the risk ranking assessment undertaken previously should be updated.

Corrosion Management Programme (CMP)

4.72. In addition to the specific RBI approach for pipework, there is also emphasis on the adoption of a corrosion management programme in accordance with the company standard across the fleet. The requirements of this standard are summarised below. It is worth noting that general compliance across the fleet is considered to be good, albeit with a couple of stations still to fully embed what is required. Actions are being developed/progressed at these stations to align their corrosion programmes with the remainder of the fleet.

- Each station shall operate an effective corrosion and degradation management programme.
- A Station Corrosion Coordinator shall be appointed at each power station.
- A Fleet Corrosion Coordinator shall be appointed.
- Each station shall prioritise their systems with baseline inspections undertaken to establish the current plant condition and identify priorities for repair, replacement and repeat inspections.

- Each station shall establish and implement an ongoing programme of regular plant walk downs by the System Engineers, incorporating inspections for corrosion and the tracking of identified corrosion.
- Corrosion awareness training shall be delivered across the fleet to the relevant stakeholders.
- Ongoing through-life oversight of fleet wide corrosion management activities shall be maintained.

4.73. In order to ensure appropriate focus is given to the corrosion management programme, a Steering Group and a Working Group are in-place, utilising the key stakeholders (i.e. Sponsoring Manager, the various Corrosion Coordinators, Fleet Engineering Standards, station Engineering Managers etc.). The Working Group presents an opportunity for each station to highlight any particular issues or concerns (both from a programme and technical perspective), including relaying operating experience arising from any inspections or from remedial activities undertaken. It is worth noting that “significant” operating experience (e.g. affecting (or potentially affecting) nuclear, environmental or industrial safety) would be documented within the company-wide Organisational Learning Portal, providing access to the information to anyone within EDF-NG.

Resources

4.74. In general, the activities discussed above will be undertaken by Station personnel, utilising specialist support as and when required (i.e. for detailed NDT, for assessment of identified degradation, etc.). To aid with the initial implementation of the risk-ranking approach taken, some stations utilised external support to minimise the impact on station staff. However, in general, the routine activities are undertaken by Station personnel.

4.1.3.2 Hinkley Point C (NNB)

4.75. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

4.76. Through-life inspection, examination, testing and maintenance will be required for safety critical pipework during the operational life of the UK EPR^{TM} as per Licence Condition arrangements. Full details of the requirements relating to Examination, Inspection, Maintenance and Testing (EIMT) of the structures will be given in the EIMT programme, which will be finalised at the end of the design process. The appropriate inspection, examination, testing and maintenance arrangements for non safety-critical pipework will be finalised at the end of the design process.

4.1.4 Preventive and remedial actions for concealed pipework

4.1.4.1 Operating reactors (EDF-NG)

4.77. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

4.78. Consistent with Section 4.1.3.1 above, it is considered appropriate to split this section into Inaccessible (or Concealed) Pipework Systems and Buried Pipework Systems, noting that there will be certain aspects (e.g. ER and defect management) that are applicable to both areas.

Inaccessible (or concealed) pipework systems

4.79. Company guidance identifies a range of remedial actions to be considered in response to adverse findings from the risk ranking and inspections undertaken.

Examples of these include pipework repair, pipework replacement, online monitoring of the system in an attempt to gauge the level of degradation (i.e. increased walk downs and monitoring of make-up tank levels (if applicable)), revising intervals between subsequent inspections and corrosion protection measures.

4.80. The measures identified within company guidance are provided as examples only and do not necessarily represent an exhaustive list. Appropriate remediation, monitoring and inspection requirements for each location shall be identified on a case-by-case basis, with input sought from relevant specialist support as required. Section 2.3 discusses the “ER Performance Monitoring” undertaken at each station. Should adverse findings be identified as part of system inspection, condition monitoring etc., a Condition Report or Work Request would be raised. This would then be subject to appropriate review at station to assess the implications and the priority of any remedial work (this would also consider potential extent of condition). The need and timescales for appropriate corrective action would then be agreed, in consultation with the appropriate stakeholders (e.g. station nuclear safety groups, central engineering support and design authority functions, etc.).

Buried pipework systems

4.81. Installed protection of the buried system provides the most effective mechanism for managing corrosion, whether internal or external, assuming that these are properly specified, installed and (where appropriate) maintained. Each of these are discussed below.

External corrosion

4.82. External corrosion is most effectively mitigated by an external coating/wrapping used in conjunction with a cathodic protection system. A coating/wrapping provides a barrier between the pipe structure and its surrounding environment and prevents the substrate coming into direct contact with the soil/backfill material. Providing that this barrier remains intact, external corrosion of the substrate should be eliminated.

4.83. Coatings/wrappings and linings (internal) are generally grouped into two categories: bonded and un-bonded linings.

- Bonded systems are defined as coatings applied to the surface (internal/external) of the pipe via a chemical/fusion process and exhibit a strong bond/adherence to the substrate.
- Un-bonded systems are mechanically attached to the pipe or expanded in-place. Similar to external coatings, wide ranges of internal linings have been used for power station plant including epoxy, cement, rubber and vinyl esters.

4.84. Where cathodic protection systems have been installed in conjunction with a coating/wrapping, this will provide the required protection should any damage/degradation occur to the coating/wrapping.

4.85. It should be noted that trial excavations of buried systems have been completed at two stations, namely Hartlepool (in 2010) and Torness (in 2011). Of the pipework inspected, it was considered to be in good condition with no significant degradation identified. Clearly, this is a small sample size across all of the buried systems across the fleet. However, it does provide some assurance over pipe condition when properly specified and installed. In general, widespread excavations are not considered appropriate, with a greater risk presented from disturbing the pipework to facilitate inspection (coupled with the potential for localised degradation to be missed from the scope of sample inspections).

Internal protection

4.86. Internal degradation is not deemed a significant issue due to the appropriate selection of pipe materials for the fluid conveyed. Linings may be applied to the internal surface of the pipework for those systems where the conveyed fluid is significantly corrosive. Such examples exist where it is not reasonably practicable to procure corrosion resistant pipework. Where pipework linings are employed, the lining is to meet the fluid duty and provide an appropriate life expectation.

4.87. Corrosion inhibitors, which are added to the system fluid, can be very effective in lowering the internal corrosion rate in closed loop systems. Biocides, surfactants and bio-dispersants can also be effective in preventing the attachment and growth of micro-organisms for raw water piping, particularly when applied to a clean pipe. Recommendations and monitoring of the use of such corrosion inhibitors is specified within TG documentation e.g. for the use of lithium hydroxide within RACW systems across the AGR fleet.

Installation of cathodic protection

4.88. Existing metallic buried pipework systems at all sites could be protected from continued external corrosion by the retrospective installation of Cathodic Protection (CP) systems. However, back-fitting a cathodic protection (CP) system to the existing buried pipe may not be practical or cost effective (any CP installed across the fleet is generally component-based (as opposed to system-based), with no buried systems currently utilising CP).

4.89. The Cooling Water (CW) system at Torness is an example of where CP has been used for some aspects of the system as follows (albeit this is not buried pipework):

- CW Intake – Stopgates and guides, coarse bar screens and drum screens.
- Pumphouse – Main CW pumps, discharge pipes and crossover pipe.
- CW Outfall – Stopgates and guides.

4.90. The requirements that must be met to ensure that cathodic protection is applied in the most economic and reliable manner are demanding. These requirements are outlined in BS EN 12954.

4.91. More recent advice has highlighted that cathodic protection has been less effective than protective coatings in mitigating Corrosion Under Insulation (CUI). Currently, there are three well established coating types for protecting against CUI – polymeric coatings, thermal sprayed aluminium and aluminium wrapping. The applications of these coatings would be considered for any repair/replacement activities, whether buried or inaccessible pipework.

4.92. As noted in Section 4.1.3.1 above, a range of monitoring and inspection techniques are identified within company guidance, with the applicability of each to be considered for the particular system, pipework section etc. These fall into four categories given below.

- Routine monitoring
- Indirect investigations
- Direct investigation (internal examination)
- Direct investigation (external examination)

4.93. As a general principle, new pipework systems or major modifications to existing pipework systems should be neither buried nor inaccessible.

4.94. Previous advice recommended that repairs and refurbishment to buried ductile iron and steel pipework should be carried out to the requirements given in BS EN 545 and CP 2010-2 respectively. BS EN 545 stresses the importance of a good external wrapping or coating for buried metallic pipework to ensure protection against corrosion damage. As outlined in BS EN 545 and CP 2010-2, various types of coating material are available and the type selected and applied depend upon the degree of protection required, the electrical resistance of the material, its resistance to water penetration, its resistance to microbiological attack and its mechanical strength and stability at the pipeline operating temperature.

4.95. As noted previously, corrosion management is currently a focus area across the fleet, with guidance on inspection recommendations, remedial actions, preventative measures, etc. developing in-line with the programme. An example of this relates to Corrosion Under Insulation (CUI) and best-practice in this area. Engineering advice has been issued, detailing the mechanism and the materials/systems at risk. In addition, guidance is provided on what inspections to undertake and what protection systems could be used to mitigate/address the risk. Similar guidance is being developed for corrosion prevention and protection.

Research and Development (R&D)

4.96. Various R&D-type activities have been undertaken to support remedial actions or modifications with regard to concealed pipework. These have either been in response to plant events or to reinforce proposed remedial activities. A number of these are discussed below.

<u>HDPE</u>

4.97. As noted earlier, HDPE pipework has been installed in a number of locations to replace vulnerable pipework. A section of HDPE pipework installed at Heysham 2 was removed after approximately four years of service. This was subjected to a series of tests and inspections to confirm the condition and to identify any degradation in-service. The report issued at the time concluded that “there is no evidence to suggest that either have suffered any deterioration in performance during the time they have been in service, which could have affected their integrity”. As a follow-up to this, it is proposed to undertake similar testing and inspection of a further section of HDPE pipework after approximately ten years of service. The overall aim here is to ensure that HDPE pipework, being installed in a number of systems at a number of stations, is performing as predicted.

<u>Puddle Pipes</u>

4.98. As discussed previously, cast iron puddle pipes at a number of stations have required remedial work or replacement. At the time, this requirement led to a review of possible repair options, ultimately resulting in the final solution implemented at the affected stations (using a carbon steel base plate and a polyurethane seal, with the seal forming part of the pressure boundary).

<u>Leak Sealing</u>

4.99. Company guidance is available for the application of on load leak sealing repairs, with this generally taking the form of leak sealing clamps or flange injections. Application of these types of repairs is common across the fleet. More recently, composite wrap repairs have been successfully undertaken (e.g. Heysham 2 Decay Heat Boiler Feed Pump sampling line and Hunterston B Back Up Cooling System

Discharge Pipework). Consideration is being given to incorporating this type of repair into the overall company guidance, providing another option to the fleet for leak sealing repairs.

Inspection Techniques

4.100. Various options for NDT inspection/monitoring of buried pipework are currently presented within the relevant company guidance document. Limitations with a number of these proposed techniques (either in terms of their application or the likely output from them) has been recognised, with the next revision of the document aiming to provide more definitive guidance on the most appropriate techniques. This will be on the basis of OPEX from attempted application of the techniques.

4.1.4.2 Hinkley Point C (NNB)

4.101. NNB advises that the appropriate design of piping and connections forms part of normal business; as such no specific preventive or remedial actions have been identified for the NAR examples.

4.2 Licensees' experience of the application of AMPs for concealed pipework

4.2.1 Operating reactors (EDF-NG)

4.102. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

4.103. IAEA have produced a programme of International Generic Ageing Lessons Learned (IGALL) reports for Nuclear Power Plants. These reports supplement the IAEA Safety Reports Series No. 82, Ageing Management for Nuclear Power Plants. The IGALL document most relevant to the concealed pipework is considered to be AMP125 (Ref 41)"Buried and Underground Piping and Tanks" (noting that tanks are not included within this discussion). Reviewing the scope and recommendations given within AMP125, it is considered that the bulk of the activities identified are already covered by the suite of documentation discussed previously (e.g. the company standard and guidance documents). This includes the implementation of suitable inspections, the use of appropriate coatings, the use of cathodic protection and trending of inspection results. As noted below, there may be elements of the company guidance (and hence AMP125) that need to be reinforced. However, it is considered that this can be addressed via the guidance as opposed to introducing a further review against AMP125 (this is considered appropriate due to the general alignment between the AMP recommendations and the existing scope of the company guidance).

4.104. A review of the success of the Inaccessible Systems Programme (ISP), incorporating the Pipework Inspection Programme, was undertaken in 2015. One of the aims of this review was to support closure of the programme of work, aiming to provide confidence that "normal business" was managing the various risks and issues.

4.105. In order to successfully close out the programme of work, it was considered necessary to demonstrate achievement of all programme requirements. As a result, programme deliverables were divided into two sub-categories:

- Non-buried pipework (including "inaccessible" and "concealed" pipework)
    - Delivery of staged RBI of pipework systems.

    - Recommendations for remedial work and future inspection strategy to support safe, reliable operation across whole fleet.
    - Identification of existing defects and subsequent interaction (e.g. production of Work Requests) with the Work Management Process to enable these to be addressed by Station.
    - Handover of data/documents/records to Station to support ongoing lifetime management of non-buried pipework systems.
- Buried pipework
    - Defined station inventories of all metallic buried pipework.
    - Delivery of 8 station based risk ranking reports, assessing metallic buried pipework.
    - Strategy for inspection of buried metallic pipework across the fleet.
    - Provision of adequate assurance of buried pipework integrity across the fleet.
    - Implementation of an asset management approach for buried pipework.
    - Handover of data/documents/records to Station to support ongoing lifetime management of buried pipework systems.

4.106. The review undertaken was generally positive, reporting a number of successes and mitigations achieved. However, gaps at a number of stations were identified, highlighting the appropriate actions for the follow-up work. Examples of this include completion of Stage 3 inspection reports, agreement on remediation plan (in response to Stage 3 inspection reports) and archiving of records.

4.107. Since the production of the review, a number of the outstanding actions have been addressed, with the remainder identified within the company's action tracking process.

4.108. It is also recognised that recent Periodic Safety Reviews (PSR) of the condition of safety related plant have been undertaken and have identified some recommendations with regard to the delivery of their ISPs. These recommendations, and their background, are discussed in more detail below.

4.109. For information, the objective of the PSR review of the actual condition of plant important to safety is to provide an assessment of the evidence that acts to substantiate the following claim: "The actual condition of safety related plant and structures are understood on an ongoing basis and will meet the design basis and functional requirements for the next PSR period. The condition of the plant is documented, and the maintenance, surveillance and in-service inspection programmes are subject to appropriate review".

4.110. As part of the ISP, Stage 1 pipework investigations at Hinkley Point B identified 18 systems as High risk, three of which were chosen to form the scope of initial inspections. From the initial inspections, external trenches were highlighted as the highest areas of risk, focussing the Stage 3 inspections on external trenches containing in-service steel pipework. Defects requiring immediate attention were repaired and a number of defects were raised to highlight issues with degradation of pipe supports, lagging and external coating. In addition, issues were also noted with the support arrangements for a number of trench covers.

4.111. The PSR review (for Hinkley Point B) concluded that: “Following this programme of inspections and repairs, there is little evidence to suggest that ongoing inspection of ‘inaccessible’ pipework was incorporated into normal business. In 2014, a nitrogen pipe failure at Hinkley Point B effectively re-opened the inspection programme, with actions to re-verify the ranking assigned to the 66 systems identified in the Phase 1 risk assessment and to carry out inspections of 16 ‘High Risk’ systems, with a particular focus on the stressors of the type which caused the nitrogen pipe failure. Following these inspections, work requests were raised to address defects, and to remove cladding and carry out further inspection where necessary”.

4.112. A PSR recommendation was raised in response to the identification of inspection of buried pipework remaining outstanding at Hinkley Point B. Inspection of trenched pipework at Hinkley Point B is currently ongoing as part of Pressure System Safety Regulations (PSSR) work, in addition to the management of systems in accordance with the corrosion management programme. The focus of the ongoing work from PSR is on the Townswater, Fixed Fire Protection and Fire Hydrant systems, based on consideration of risk, with a feasibility study to be undertaken to confirm the most appropriate risk mitigation activities.

4.113. The Hunterston B Stage 1 risk assessment identified 30 pipework systems that were considered as high risk (i.e. requiring further investigation in Stage 2). The main conclusion from the risk ranking was that external trenched pipework had the highest risk profile. Accordingly, Hunterston B carried out inspections of all trenched pipework external to the Reactor Building and refurbishments were carried out in 2011.

4.114. The PSR review (for Hunterston B) identified that inspection of trenched pipework within the Reactor Building remained outstanding. Much of the pipework is either safety related or poses a potential hazard to other safety related equipment. There are also environmental implications as well as general reliability issues. A PSR recommendation was raised for station to develop a programme for inspection of outstanding at-risk trenched pipework and ensure that identified defects are prioritised and incorporated into normal business, in addition to the development of a programme of more regular inspections. These pipework inspections have been incorporated into the ongoing Corrosion Management Programme.

4.115. The equivalent Heysham 1 and Hartlepool PSR document raised a similar recommendation, judging that further work is required to effectively implement the TG issued on corrosion management. There remains ongoing focus in this area from a fleet perspective, aiming to ensure a consistent and suitable approach is taken to corrosion management at all of the stations. Both stations have also reported incidents involving inadequate external protective coating standards on pipework. This has the potential to lead to corrosion and further degradation of the pipework in service. As a result, the PSR document has raised recommendations to address these issues.

4.116. Based on these PSR findings, it is considered likely that the equivalent PSRs undertaken at the other stations across the fleet will identify similar recommendations. A mitigation to this is the increased focus and priority being assigned to corrosion management, primarily in response to a number of in-service failures experienced. It is considered that the focus being given to corrosion management across the fleet, supported by relevant TG documents, should ensure that any similar shortfalls at other stations are addressed without the need to rely on PSR to identify any shortfalls.

4.117. The focus on corrosion management (both from internal and external stakeholders) should also help to ensure that appropriate inspections of the relevant systems are being undertaken, with any identified adverse findings being addressed in a timely manner (ranging from repeat inspections, minor remedial work, replacement etc.).

4.118. It is recognised that defects may be identified in the future. However, the steps being taken to embed corrosion management across the fleet should ensure that these defects are identified and suitable remedial steps taken in a timely manner to minimise the likelihood of a significant defect occurring.

4.119. In addition, it is recognised that monitoring and repair techniques and technologies are evolving, with the need to keep awareness in these areas. Fleet forums are in-place to highlight developments in these areas, with consideration (as required) for review and endorsement from a fleet perspective. An example of this is the advice note issued on coatings to prevent corrosion.

Recommendations by EDF-NG

4.120. This review, in conjunction with the overall programme of TG production, has highlighted the benefit in amalgamating some of the existing guidance on system risk-ranking and inspection. This will be considered as part of the routine review and update of these documents, which will be issued by 30 June 2018.

4.2.2 Hinkley Point C (NNB)

4.121. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

4.122. EDF-NG’s fleet has considerable experience in the application of AMPs for concealed pipework. Findings are shared across the fleet to ensure a better understanding of ageing issues. Participation of NNB in EDF-NG’s ‘Corrosion Management User Group’ allows NNB GenCo to understand and monitor corrosion and ageing management issues faced by EDF-NG’s fleet and facilitates applying this operating experience to Hinkley Point C.

4.3 Regulator’s assessment and conclusions on ageing management of concealed pipework

4.3.1 Criteria and standards for assessing ageing management

4.123. The relevant criteria and standards adopted within ONR’s assessment are principally our Safety Assessment Principles (SAPs), ONR Technical Assessment Guides (TAGs) and ONR Technical Inspection Guides (TIGs), together with relevant national and international standards and relevant good practice informed from existing practices adopted on UK nuclear licensed sites. In addition to the general criteria and standards described in Section 2 of this report, ONR has utilised the following in the assessment of the licensee’s arrangements for ageing management of concealed pipework:

- AMP125 – Buried and Underground Piping and Tanks, International Generic Ageing Lessons Learned (IGALL) (Ref 41)
- RR823 – Managing Ageing Plant, A Summary Guide, Health and Safety Executive (HSE) (Ref 42)
- IGC Doc 190/14 – Plant Integrity Management, European Industrial Gases Association (Ref 43)

- 1007933 - Ageing assessment field guide, Electric Power Research Institute (EPRI) (Ref 44)

4.124. Section 4.1.1.1 describes the scope of what EDF-NG considers to be ‘concealed systems’. This scope is wide ranging, covering all designs of pipework where access to the external surface of pipe is not readily available. ONR notes that EDF-NG’s interpretation of concealed systems is in excess of the concealed pipework designs described in the technical specification for the NAR (Ref. 1) and is supportive of this wide breadth consideration of pipework designs where opportunity for in-service inspection is limited. In accordance with Ref. 1, ONR’s assessment has only targeted EDF-NG’s ageing management approach for pipework designs which are buried in soil or concrete, cast in concrete or located in covered trenches.

4.3.2 ONR’s assessment of EDF-NG’s ageing management of concealed pipework

4.125. ONR has reviewed a sample of the technical guidance and standards documentation referenced within the NAR and drawn upon the existing experience gathered from ONR’s regulatory engagements with EDF-NG, which are relevant for the ageing management of concealed systems

4.126. IAEA guidance on ageing management for nuclear power plants refers to a systematic approach to managing ageing of a structure or component (Understand, Plan, Do, Check, Act). To be most effective, this approach needs to be given consideration at the design stage, especially when considering any system where the scope for subsequent inspection is limited, such as for concealed pipework.

4.127. As described above in section 4.1, EDF-NG’s approach to ageing management of concealed pipework is a combination of plant management activities (driven by safety case or regulatory requirements to produce operating rules, instructions and maintenance schedules) and inspection programmes developed and implemented in accordance with technical governance arrangements to bring about a desired improvement.

4.128. Through consideration of the Plan stage of the cycle, retrospectively developing and optimising activities for ageing management of concealed pipework is challenging, especially when dealing with plants that have been operating for over 30 years. EDF-NG has implemented arrangements through its TG process, which has resulted in a number of targeted programmes to coordinate, maintain and improve activities for ageing management. ONR judges that EDF-NG has demonstrated this has been achieved for a number of activities that have been conducted for the NAR examples.

4.129. For concealed pipework, the Do stage of the cycle is within EDF-NG’s arrangements and is predominantly linked with the nuclear safety function of the SSC. Operation of the pipework though temperature, pressure or chemistry control is maintained through compliance with the nuclear safety case requirements, and managed through technical specifications and procedures supported by relevant maintenance schedules. Evidence has been noted through ONR regulatory site inspections of plant operating history (e.g. jockey pump monitoring of buried fire-fighting systems, header tank make up rates) being used to monitor trends as a way of providing confirmation of pipework integrity.

4.130. EDF-NG’s approach to improve ageing management and plant reliability for the NAR examples for the Check stage of the cycle has predominantly been through the inspection, monitoring and assessment of SSCs, implemented as targeted programmes (i.e. ISP, CMP) through the TG process.

4.131. From consideration of the NAR examples, EDF-NG's approach has demonstrated that correction of pipework degradation and design modifications have been implemented in accordance with the Act stage of the cycle. An example of this is the changes to component design of buried safety significant cooling water pipework (such as replacing buried cast iron pipework with over-ground glass flake lined carbon steel/HDPE pipework) which has been implemented by EDF-NG at considerable cost.

4.132. As part of the ONR's routine regulatory site inspections, it regularly sample the licensee's arrangements for conducting safety case or regulatory-driven EIMT activities. ONR has reviewed a sample of recent ONR regulatory inspection and assessment reports relevant for the NAR examples which are described below. EDF-NG's management of plant material condition of pipework (including trenched and buried systems) is currently the subject of a targeted ONR intervention outside of its routine regulatory inspections (described in more detail below), which has provided a broad source of inspection and assessment information to support ONR's judgement and conclusions presented within this report.

4.133. The technical guidance and standards utilised by EDF-NG contains detailed information on how to devise and implement an effective RBI strategy to determine the material condition of pipework. ONR considers these documents to be informative and sourced from appropriate references, which include industrial and international standards and guidance.

4.134. ONR has reviewed the three guidance documents described in section 4.1.1.1, and is satisfied that they are aligned with the fundamental principles of a risk based inspection programme, as presented within the criteria and standards listed in sections 2.1 and 4.1 above.

4.135. Overall, ONR considers that EDF-NG's processes and technical guidance documents described within section 4.1, utilising a risk based inspection approach, account for the scope of concealed pipework which is buried in soil or concrete, cast in concrete or in covered trenches. From the sample regulatory inspections undertaken as part of ONR's intervention on buried systems, the majority of routine EIMT activities conducted on concealed pipework important for safety are judged to be done so in accordance with the technical guidance and standards described. Where this hasn't been demonstrated, ONR has rated the outcome of the regulatory inspection in accordance with our internal guidance for inspection, and taken appropriate enforcement action to address the identified shortfall against expectations.

4.136. Section 4.2.1 explains how EDF-NG has applied its experience of ageing management through delivery of targeted inspection programmes. As explained in section 4.1.1.1, the launch of these inspection programmes and production of subsequent technical guidance documentation was in response to a number of significant system failures, which revealed shortfalls in the ageing management approach in use at the time.

Further Work Being Undertaken by EDF-NG to Improve the Effectiveness of Concealed Pipework Ageing Management

4.137. As part of the ongoing CMP, EDF-NG is currently in the process of revising the technical guidance on the management of trenched and buried pipework important for safety, to integrate it into the overarching CMP across all sites. This will enable the ageing management of trenched and buried systems to be managed under the company technical standard on corrosion management, consolidating resource and maintaining consistency of approach. The CMP aims to determine the existing material condition of pipework through a risk-based approach, conduct any necessary remedial work and implement an appropriate forward inspection strategy commensurate with the safety function of the SSC and other existing EIMT arrangements.

4.3.2.1 ONR's Experience from regulatory site inspection and assessment as part of its regulatory oversight

4.138. As discussed previously, ONR regularly assesses the adequacy of EDF-NG's routine EIMT activities for SSCs important for safety through normal regulatory business. Through these engagements, ONR has sampled a number of SSCs within the NAR examples as described below. The development and implementation of EDF-NG's recent corrosion management strategy is the subject of a targeted ONR intervention, which is described later in this chapter.

4.139. As part of its normal regulatory business, ONR conducts System Based Inspections (SBIs), as discussed in para. 2.226. The purpose of an ONR SBI is to assess whether the licensee has implemented adequate arrangements to fulfil the fundamental safety, operational and maintenance requirements of the safety case.

4.140. A number of the concealed pipework groups falling within the NAR specification have been sampled in these SBIs in recent years. A sample of these SBIs has been reviewed for systems that contain concealed pipework relevant to the identified NAR examples. From the sample review, ONR has not raised any significant concerns with respect to EDF-NG's existing approach to managing EIMT activities for buried or trenched pipework.

4.141. Therefore, ONR is broadly satisfied that, from the information sampled under the SBI programme, no significant issues related to the management of concealed pipework have been identified for the NAR examples.

4.142. Through ONR's normal regulatory engagements with EDF-NG, structural integrity assessment inspections are carried out during every periodic shutdown of a reactor site (typically every three years at AGR sites).

4.143. The puddle pipes described in section 4.1.1.1and 4.1.4.1 are an example of safety significant pipework that is encased in concrete. These puddle pipes have been sampled as part of these structural integrity periodic shutdown assessments, with no issues related to the management of puddle pipe inspections identified. From this sample, ONR is therefore broadly satisfied that EDF-NG's approach to managing ageing of safety significant pipework encased in concrete in accordance with their internal technical guidance is satisfactory.

4.144. EDF-NG has recently (2016) completed a periodic safety review for Hinkley Point B and Hunterston B safety cases. The ONR structural integrity assessment of this PSR included a review of EDF-NG's strategy for managing buried and trenched systems. ONR concluded that in general, EDF-NG has good arrangements in place for ageing management, however these had not yet been fully implemented at these sites. A recommendation was therefore made for ONR to continue monitoring EDF-NG's progress through a targeted intervention on corrosion management of pipework, as discussed above.

ONR Targeted Interventions

4.145. In response to recent pipework failures associated with corrosion under insulation (CUI), ONR launched a fleet-wide intervention on corrosion management to assess the adequacy of EDF-NG's arrangements and how effectively they have been implemented at each site, including a review of EDF-NG's strategy for managing buried pipework.

4.146. As discussed previously, ageing management of concealed pipework across EDF-NG sites is delivered through a programme of long-established proactive safety-case/regulatory identified EIMT activities and reactively implemented RBI programmes in response to emergent plant failures or identified shortfalls in existing arrangements.

4.147. In ONR's opinion, the effectiveness of EDF-NG's approach to ageing management of concealed pipework was not easily demonstrable, making it vulnerable to 'gaps' in understanding of material condition. Where this has become apparent as a shortfall (e.g. failure to identify pipework CUI), EDF-NG has generally been effective in implementing a recovery strategy for managing assessment of the extent of condition of those SSCs affected. This has been governed and guided by technically relevant standards and guidance that is judged to be in line with applicable industrial standards and guidance.

4.148. ONR notes that EDF-NG is progressing a significant programme of fleet-wide inspections and remediation activities associated with the ageing of concealed pipework relevant to the NAR examples. This was necessary as a result of the shortfalls identified in the effectiveness of its management of concealed pipework. These activities are generally being carried out in accordance with the technical guidance and standards provided, which are supportive of an effective ageing management programme.

4.149. ONR notes that EDF-NG has implemented a steering and oversight function consisting of technical and safety representatives, which feed into central oversight functions for safety and engineering. Underneath the central oversight and delivery function, EDF-NG has appointed a working group consisting of central SQEP engineers (Suitably Qualified and Experienced Persons, a licence requirement) and site-specific corrosion coordinators, who implement the CMP locally with support from its inspection and maintenance staff. These governance and oversight functions meet on a regular basis, and ONR has observed a good level of senior management support and sponsorship at a site level. ONR considers that EDF-NG's general approach to training and raising awareness of ageing management for concealed pipework is good, having improved significantly over the last two years. Under the CMP, a large group of individuals have received training to maximise effectiveness of EDF-NG's inspection and defect recording programmes.

4.150. ONR is broadly satisfied that EDF-NG has demonstrated that the mandatory requirements of its technical standard on corrosion management are being delivered centrally and at each of the sites, as presented in section 4.1.3.1. However, ONR has identified that there are a number of areas for improvement with respect to the consistency in approach to the implementation of these arrangements on some sites. The areas for improvement are predominantly associated with the management of defect remediation activities in response to EDF-NG inspection findings. Recording of judgements made with respect to remediation prioritisation also requires improvement for a number of sites. These shortfalls are the subject of ongoing ONR engagement with EDF-NG through targeted interventions. ONR has observed that EDF-NG has responded appropriately to OPEX gathered from the implemented corrosion inspection programme, which has been fed back into the technical guidance and standards on inspection and remediation strategies.

Pipework Located in Covered Trenches

4.151. EDF-NG's ISP launched an inspection strategy lasting several years, which targeted trenched pipework, utilising the risk based inspection (RBI) approach. This approach is specified within a number of the international standards and criteria considered within this review, the fundamental principles of which are embedded within EDF-NG's technical guidance documents.

4.152. ONR's regulatory inspections at Heysham 1 and Hartlepool sites have identified examples where EDF-NG has inspected and completed remediation of trenched pipework, including safety significant fuel transfer pipework for emergency power generation and transfer of essential cooling water. ONR is satisfied that this evidence demonstrates that EDF-NG is putting into practice advice on remediation and inspection strategies presented within the relevant company technical guidance for the NAR examples.

Buried Pipework

4.153. In accordance with the technical guidance for managing buried pipework, EDF-NG has implemented a RBI strategy for buried pipework, which has resulted in the production of site specific buried pipework risk ranking reports that identify sections of pipework considered to be most at risk of corrosion.

4.154. From the sample of cases reviewed in our regulatory inspections, it is apparent that most of the work completed was in accordance with the technical guidance on buried systems for risk ranking the pipework, which was completed during the ISP when it was predominantly under the control of EDF-NG's fleetwide projects group. Following completion and close out of the fleetwide ISP, responsibility to implement later phases of the technical guidance for developing ongoing maintenance regimes and remediation strategy for buried systems, did not fully align with the advice given in the technical guidance. This finding aligns with the issues identified by EDF-NG in Section 4.2 and as such, EDF-NG is already taking steps to review these issues and provide support to the sites to revise the current arrangements for managing buried pipework to integrate it with the existing company technical standard on corrosion management. To confirm that the existing arrangements are adequate against the licence condition requirements for EIMT, ONR sampled buried pipework arrangements for a number of systems identified in the risk ranking report at both Heysham 1 and Hunterston B.

4.155. EDF-NG was able to demonstrate a good understanding of the condition of the back-up cooling water buried pipework, gained from opportunistic confirmatory checks completed during an unrelated plant modification. A review of the radioactive effluent pipework on site confirmed that appropriate EIMT pressure test routines are being conducted to confirm pipe integrity every two years. It has been noted from ONR's interactions with Hunterston B as part of our intervention on corrosion management that the site has a considerably more advanced programme for managing corrosion than at other sites. Therefore these results may not be indicative of practices within EDF-NG more generally, and further sampling of EDF-NG's arrangements for managing buried systems at other sites is considered necessary.

4.156. In ONR's opinion, EDF-NG has demonstrated that the sampled sites are applying company guidance on the management of buried pipework, which includes evidence for appropriate EIMT activities and taking full advantage of opportunistic inspections where available. ONR and EDF-NG have nevertheless identified a number of shortfalls related to the sustained implementation of inspection and remediation activities advised in the technical guidance. These shortfalls are in the process of being addressed by EDF-NG, and we will continue to monitor progress through our fleetwide intervention on corrosion management.

4.3.3 ONR's assessment of NNB's ageing management of concealed pipework

4.157. In undertaking this assessment, ONR considered the current status within the reactor site lifecycle and the current level of maturity of the ageing management programme.

4.158. From the information reviewed, it is apparent that the design process for the UK EPR has given good consideration to the accessibility of pipework important for safety, and as such only a minimal number of SSCs contain pipework that is concealed under the definition provided in the specification (Ref 1). Where concealed pipework does exist, NNB advises that ageing management has been accounted for in the design through an understanding of the assigned safety function of the pipework and the predicted degradation mechanisms.

4.159. Currently, there are no routines or strategies in place for managing EIMT activities of concealed pipework, on the basis that this will be developed and implemented to comply with the requirements of the licence condition for EIMT closer to the date of commissioning. As a result, this limits the basis for which ONR can make any assessment or judgement on NNB's approach to ageing management for the UKEPR. It is ONR's expectation that, although there are currently no routines in place for the UK EPR, NNB should have considered and reported on relevant OPEX from other EPR operators who are nearing completion of construction or plant operation. This would provide for proactive consideration of what ageing management programmes are in place, so that a comparison could be made between NNB's AMPs and ONR's regulatory expectations.

4.3.4 Overall conclusions on ageing management of concealed pipework

4.160. ONR has assessed EDF-NG's arrangements for ageing management of concealed pipework as presented within its central technical guidance and governance documentation, with further evidence gathered from site-based regulatory inspections that have been conducted as part of ONR's normal regulatory activities and other work completed for this review. Based on this, ONR considers that EDF-NG's arrangements contain the necessary elements of an effective ageing management programme.

4.161. ONR's recent intervention has highlighted that EDF-NG's historic concealed pipework inspection programmes were inadequate, and these shortfalls were recorded on the ONR regulatory issues database, with the requirement to seek regulatory enforcement action to secure the necessary improvements. Through the duration of ONR's ongoing intervention and oversight through the open regulatory issue, EDF-NG has demonstrated an increased understanding of material condition, which has resulted in significant plant modifications, including repair and replacement of safety significant concealed systems, in some cases modifying the pipework structure so that it is no longer concealed. ONR considers this to be indicative that where implemented, EDF-NG's strategy for remediation has proactively considered appropriate factors for ongoing inspection and maintenance which is aligned with relevant standards and guidance for ageing management of concealed pipework.

4.162. Whilst ONR judges that EDF-NG's approach to ageing management of concealed pipework is broadly aligned with international standards, it considers that there are still areas for improvement with respect to the fleetwide strategy for managing prioritisation of remediation activities and consistency in the approaches to pipework inspections. ONR will continue to maintain regulatory oversight of EDF-NG's management of concealed pipework until EDF-NG can provide satisfactory evidence to demonstrate that the shortfalls identified by ONR have been addressed adequately. EDF-NG acknowledges this and is openly engaging with ONR to support the resolution of issues raised during our regulatory inspections (where shortfalls have been identified with respect to forward planning of remediation activities and corporate accountability for site-comparative CMP progress). EDF-NG is also reviewing its existing buried pipework inspection strategy and updating its governance and technical guidance documentation to align and consolidate it within the existing fleetwide CMP. The latter has been included as a finding of this report, with a completion date of 31 December 2018.

4.163. To ensure that these improvements are implemented in a timely manner they have been brought together along with those from other chapters in Chapter 9. Chapter 9 identifies a single area for improvement (AFI) for EDF-NG to undertake a programme of improvements and the individual elements necessary to complete that programme are clearly identified, along with dates for their completion.

4.164. ONR is also currently conducting an intervention to review how EDF-NG has been managing corrosion of concealed pipework, specifically targeting buried pipework inspection and maintenance strategy at each site, with the next fleetwide progress update planned for March 2018. The areas for improvement identified within this NAR are directly related to ONR's existing intervention on corrosion, and as such will be discussed during this interaction and managed in accordance with the existing regulatory engagement.

4.165. For NNB, ONR considers that the current level of maturity of the company's ageing management programme for concealed systems is not yet developed sufficiently to identify specific strengths or weaknesses. ONR recognises that consideration for ageing and inspection issues associated with concealed pipework that have been applied during the design stage, has minimised the amount of concealed pipework on the UK EPR design. This approach is judged to be supportive of an effective ageing management process, and is considered to be an effective preventative measure to minimise and control ageing degradation through proactive consideration of factors that may inhibit material performance, inspection and assessment through operational life. From a concealed pipework perspective, ONR supports NNB's proposals for the preparation of a corrosion and ageing management strategy as identified in Section 2.7.4.

4.166. No specific areas for improvement have been identified for Hinkley Point C.

4.167. The overall conclusion from the assessment of concealed pipework is that both licensees have adequate AMPs, given the specific stages of the lifecycles of their plans, and that EDF-NG needs to make a limited number of secondary beneficial improvements.

5 Reactor pressure vessels

5.1 Description of ageing management programmes for RPVs

5.1.1 Scope of ageing management for RPVs

5.1.1.1 Operating reactors (EDF-NG)

5.1. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

5.2. The continuous safety of the Reactor Pressure Vessel (RPV) is ensured by compliance against the arrangements set out in the Nuclear Site Licence Conditions.

5.3. EDF-NG operates a single PWR power station in the UK at Sizewell B. The following section is not relevant to the EDF-NG AGR fleet.

5.4. The RPV at Sizewell B consists of four large forged components, a Core Shell, a Nozzle Shell Course, a Bottom Dome and a Top Removable Dome (a more specific component listing is provided below). The top dome (RPV Closure Head) has penetrations to allow reactor control, while the bottom dome has instrumentation penetrations to monitor reactor neutron flux (see Figure 5.1).

5.5. The RCS is based on the Westinghouse Standardised Nuclear Unit Power Plant System (SNUPPS) design and the RPV was designed in accordance with ASME III which remains one of the industry standards for light water reactors.

5.6. During the consenting and design process additional features were included in the design in order to satisfy the requirements of the Public Enquiry and the Government's Chief Scientific Officer. These requirements included justification of certain components to a level of integrity such that failure was deemed incredible. A figure of 10^{-7} per year was attributed to this and the overall concept called Incredibility of Failure (IoF). At Sizewell B the RPV is an IoF component.

5.7. All of the elements that make up an AMP for the Sizewell B RPV are either embedded in the station safety case (managed by the MDI process) or are part of company procedures and processes, see Section 2 for the overall description of these activities and processes. The safety case specifies the monitoring, inspection, testing and maintenance programme, which includes ISI designed to monitor and manage plant degradation. These are captured by the surveillance programme (which is analogous to the MS for the AGRs), which sets out the frequency of RPV examinations and other RPV AMP activities. The focus of the commentary in Section 5 is therefore on the surveillance arrangements for the ageing management of the RPV.

5.8. Due to the passive nature of the RPV, the AMP surveillance and condition trending activities do not strictly align with the company ER processes. The ER process is focused predominantly upon plant and system components where active changes in plant configuration & performance can be readily measured and recourse to condition monitoring methods can be used to generate informative metrics. The long term condition and trending of the pressure vessel is included in the existing SHIP processes but the output offers only a limited overview of the actual structural performance and understanding required to provide relevant information to inform forward AMP strategies. Instead, the long term performance of the RPV is more effectively informed by the detailed surveillance programmes and associated reporting.

5.9.

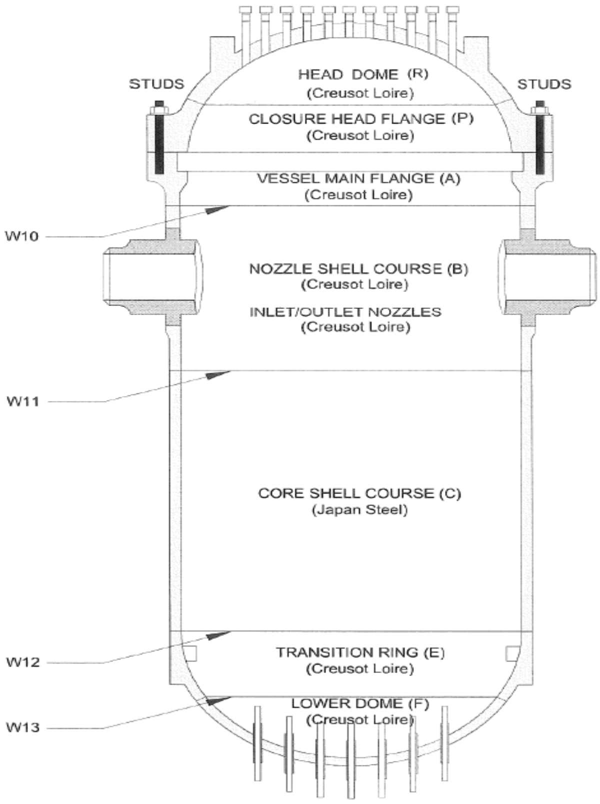


Figure 5.1 – General arrangement of showing where RPV forgings were manufactured.

5.10. The RPV is subject to a number of company TG standards establishing the governance arrangements through mandatory requirements to control activities and safety. The following technical standards are applicable to the RPV:

- **Corrosion management (mechanical).** This company standard establishes the requirements for implementing and improvement of an effective approach for corrosion management.
- **Controlling the quality of non-destructive testing.** In-service inspections through non-destructive testing techniques are governed by this company standard in order to ensure consistency and quality. Inspections by EDF-NG of the RPV fully comply with the arrangements.
- **Pressure systems.** The purpose of this document is to set the requirements for the management of all components that contain liquids at pressure. These requirements are considered mandatory in order to meet the objectives of safe and reliable operation and to comply with relevant legislation, such as Pressure System Regulations.
- **Primary chemistry optimisation plan and primary water chemistry specification.** This standard sets out the chemistry requirements for the primary circuit. The specification follows the requirements as set out in the EPRI Guidelines.

- **Management of Structural Integrity Branch outage related inspection and assessment activities.** This guidance provides the management arrangements for inspections and assessments during an outage. In the case of the RPV this covers how the inspection results are assessed and reported. It should be noted that some of the activities are under the control of the EDF-NG Structural Integrity Panel (SIP).

5.11. Arrangements also exist for the management of changes in data or methods that may affect existing safety cases. This is applicable in the case of the fracture analysis that underpins part of the structural integrity of the RPV. The development of fracture assessment codes and assessment methodologies is controlled by the EDF-NG Structural Integrity Steering Board and the EDF-NG Structural Integrity Review Committee.

5.12. The original design life for Sizewell B is 40 years. Given the quality of build and the conservative design features it is likely the plant can be justified for an extended period of operation beyond this. The processes used to manage plant life extension (see paragraph 2.48), have not been deployed for the Sizewell B RPV at this stage of its operating life, however, work is due to commence in the near future.

Scope of ageing management for RPVs

5.13. The RPV is made from SA508 Class 3 ferritic steel with layers of stainless steel cladding. The RPV Closure Head was replaced in Refuelling Outage 8 (RFO8) in 2006 and incorporates Alloy 690 penetrations and associated weld metal, which is now common in industry and is considered to be significantly less susceptible to stress corrosion cracking. Due to physical constraints and operating temperatures the bottom dome remains with bottom mounted instrumentation (BMIs) penetrations made from Alloy 600 and Alloy 82/182 weld material. The RPV forgings were made by two manufacturers. The RPV Core Shell was made by Japan Steelworks (JSW), while the remainder of the RPV was made by Framatome, now Areva at Creusot-Forge. See Figure 5.1 for details. The whole structure was welded together at the Framatome works (St Marcel).

5.14. Manufacture of the RPV began in the late 1980s and prior to shipping to Sizewell B was subjected to a hydro static test. A further hydro test was conducted once the vessel was installed and connected to the rest of the Reactor Coolant System (RCS). The RPV has a design pressure of 17.2 MPa, which is 1.1 times the normal operating pressure of 15.5MPa. Two hydro tests were performed at 21.5MPa.

5.15. In order to meet the company's IoF requirements (see paragraph 5.6) additional measures beyond those required under ASME were adopted. This has included detailed fracture assessment based on a failure assessment criterion that takes account of both ductile and brittle failures, also inspection that are classed as validated based on independently verified blind trials covering inspectors, procedures and equipment.

5.16. The RPV safety case is based on two main legs, the Achievement of Integrity and the Demonstration of Integrity. The Demonstration of Integrity takes the Validated Inspection Defect Size and conservatively assumes that is the size of a postulated start of life defect size. This is then subjected to a fatigue crack growth assessment using predicted life time cycles to give an end of life crack size, which in turn is then used to determine a margin of safety called a Validation Factor and is compared with the size of defect (limiting) that is predicted by analysis to be of concern to safety. Hence:

$$\text{Validation Factor} = \frac{\text{Limiting Defect Size}}{\text{End of life Defect Size}}$$

5.17. The safety case requires that all Validation Factors be at least ‘approaching 2’. It should be noted that there are further pessimisms in the RPV design and fracture analysis providing additional conservatisms. In order to take account of through life operations the design assumes a conservative number of thermal and pressure transients.

5.18. The main structures and components that constitute the pressure boundary are shown in Figures 5.1 and 5.2 and consist of the following key components:

- RPV Closure Head (RPV Closure Head Flange, Closure Head Dome and Control Rod Drive Mechanisms (CRDM) penetrations)
- RPV Flange (including RPV Head Seal and RPV Head Studs)
- Nozzle Shell Course (including 4 off Inlet Nozzles and 4 off Outlet Nozzles)
- RPV Core Shell
- RPV Transition Ring
- RPV Lower Dome (including bottom mounted instrumentation penetrations)

5.19. For convenience the RPV components have been separated out into three groups and the main ageing mechanisms considered for each group. The following details the applicable ageing mechanism that Sizewell B has identified.

5.20. RPV Core Shell including base metal, cladding and welds (Group 1)

- Irradiation embrittlement of RPV welds and base metal
- Thermal ageing effects of weld and base metal
- Under clad cracking
- Fatigue mechanisms

5.21. RPV Head and Lower Dome including penetrations and Studs (Group 2)

- Stress Corrosion Cracking of Alloy 600 penetrations
- Fatigue mechanisms
- Under clad cracking
- General corrosion of Studs

5.22. RPV Nozzle Shell Course, Inlet and Outlet Nozzles (Group 3)

- Fatigue mechanisms
- Under clad cracking

5.23. The methodology and criteria for selecting components within the scope of ageing management has developed over many years and is enhanced through international operating experience (OPEX).

5.24. For the first group (Group 1) of components irradiation embrittlement is arguably the most important as this includes sections of the pressure vessel closest to the reactor core and hence subject to this ageing mechanism. Sizewell B has a comprehensive materials programme to monitor irradiation embrittlement and thermal ageing, embedded in a surveillance programme. This programme does not consider other components that make up the RPV as the dose subjected to the RPV Core Shell bounds that of the other RPV forgings as the irradiation effects have been confirmed to be significantly less.

5.25. The inspections capability deployed at Sizewell B is capable of detecting cracks at the cladding/pressure vessel interface. The safety case makes no claims on the cladding. Under clad cracking also applies to the other RPV components listed under the other two groups.

5.26. Fatigue is managed under a transient management programme. It considers all the RPV components as a whole. The programme records pressure, temperature, flow variations and load changes to establish how many transients have accrued. These are then compared with the lifetime allowable for 40 years of operation. There is no actual measure of fatigue damage to the RPV, however the periodic reviews (monthly and annually) provide a clear picture of the RPV status when set against the design schedule. EDF-NG is currently investigating whether an on-line fatigue management programme is required. Details of fatigue monitoring are given below.

5.27. For the RPV Head and Lower Dome penetration (Group 2) the principle degradation mechanism is considered to be stress corrosion cracking (SCC). This is due to the material choice for the penetrations and welds. At the start of life both domes had penetrations made of Alloy 600 with the equivalent weld materials used. Following industry concerns, in particular the Davis Besse incident, the RPV Closure Head was replaced using the Alloy 690 penetrations base material and weld equivalent Alloy 52. Alloy 690 is considered to be significantly less susceptible to SCC. The work to replace the RPV Closure Head was completed in 2006.

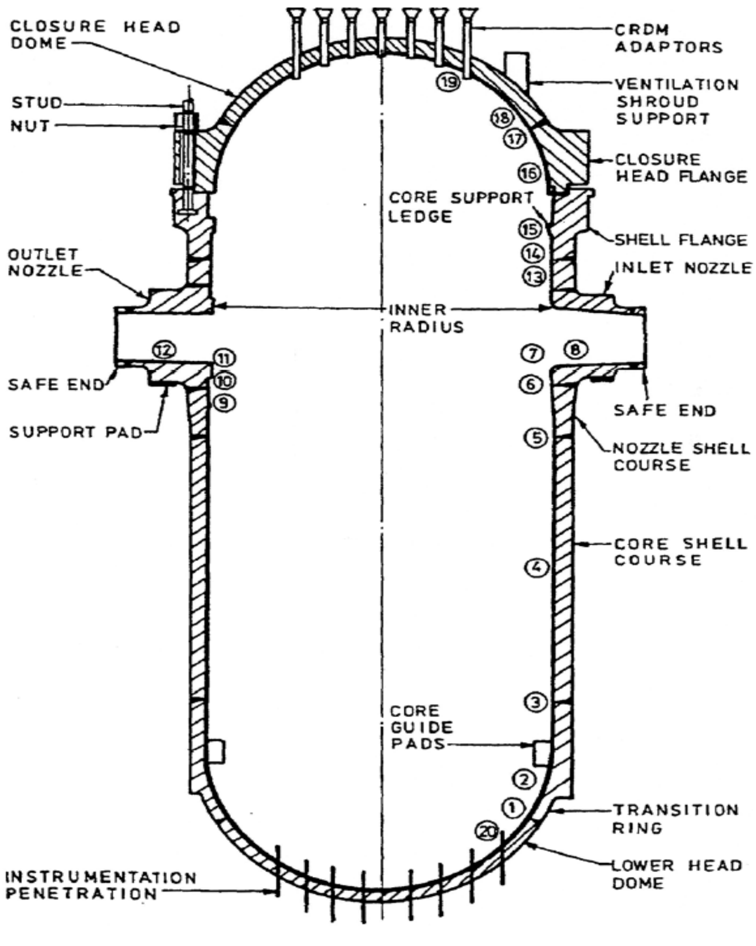


Figure 5.2 - General Arrangement of RPV Showing Locations of Fracture Analysis

5.28. As it is not practicable to replace the RPV Lower Dome the inspection regime has been changed. The Lower Dome is also considered to be less susceptible to SCC on the basis of the operating temperature, below the SCC initiation temperature. Details of the changes to the inspection regime are given later.

5.29. EDF-NG has considered the threat from SCC in general and taken steps to monitor and mitigate the mechanism. This includes regular OPEX reviews under a surveillance programme (a frequency of once per period typically every 3-4 years) and through R&D collaborations e.g. Electrical Power Research Institute (EPRI) and Pressurised Water Reactor Owners Group (PWROG).

5.30. The RPV Closure Head Studs are inspected in line with the ASME XI code and replaced if required.

5.31. The components listed under Group 3 are covered by the same activities above.

5.1.1.2 Hinkley Point C (NNB)

5.32. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

5.33. The HPC nuclear power plant is an EPR^{TM} design, and the RPV design is based on the principles and equipment already implemented in French N4 plants. The design Code used for the RPV is RCC-M, and the design life is 60 years. The RPV is the pressure boundary for the core and comprises two main components connected by closure components. These are the reactor vessel body, and the vessel closure head. All the internal surfaces of the RPV are clad with stainless steel.

5.34. The reactor vessel body consists of the following sections (see Figure 5.3), welded together by circumferential welds. The upper part is a single unit comprising a nozzle shell with an integral flange. The nozzle shell includes eight penetrations to connect four loops of the reactor coolant system pipework. The lower part of the reactor vessel body is made up of two core shells, a transition ring, and a lower head. These are all forged and are welded together by circumferential welds. The two cylindrical core shells encompass the active height of the core and are free from discontinuities.

5.35. The vessel closure head is made up of two welded components:

- The vessel flange is a forged ring with holes for the closure studs. The lower face of the flange is clad with stainless steel which is grooved to form the recesses for two head seal gaskets.
- The upper head is a forged component, partially spherical in shape, penetrated by shrink-fitted and welded tubes which allow access to the Rod Cluster Control Assemblies (RCCA) drives and instrumentation.

5.36. During the Generic Design Assessment (GDA) process, the RPV was classified as a High Integrity Component. For these components, a specific set of measures are implemented to achieve and demonstrate their integrity, as gross failure of such components is considered to be outside of the design basis for HPC. There is a three-part approach applied for minimising the risk of fracture, as follows:

- Use of fracture mechanics to determine the End of Life Limiting Defect Size (ELLDS), and Lifetime Fatigue Crack Growth (LFCG) from a Qualified Examination Defect Size (QEDS) in order to determine a Defect Size Margin. The target is to achieve a ratio of ELLDS to LFCG+QEDS approaching two.

- Use of suitable redundant and diverse inspections during manufacturing, supplemented by the use of qualified inspection(s) at the end of manufacturing. Repeat inspections are planned of all RPV forgings and welds to give additional confidence in freedom from structurally significant defects.

- Verification of the lower bound fracture toughness values used to determine the critical defect size by measurements (Fracture Toughness Testing)

5.37. The verification of the lower bound fracture toughness values in the ductile range used to determine the critical defect size will be performed for the base metal by measurements of the fracture toughness on RPV forging prolongations, and for the weld material on a mock-up. The mock-up will comprise the same filler wire/flux combination as the RPV core shell weld and the other RPV welds.

5.38. Low-alloy ferritic steel is used for the RPV shells, flanges, transition ring, nozzles, and upper and lower hemispherical domes. This material has been used extensively on previous French PWRs. In the active area of the core subject to high levels of irradiation (core shells), the copper and phosphorus contents are limited to mitigate the effects of embrittlement by irradiation. The cobalt content is also limited which is related to reducing the level of radiation on the vessel interior wall. A very low content of residual elements is also required in order to obtain the right properties for toughness and good weldability. Although no limit is explicitly defined in RCC-M, very low contents of antimony, tin and arsenic are required for all RPV forgings.

5.39. Although limits set in RCC-M for residual elements in welding consumables are higher than those specified for base materials AREVA's internal database shows that the actual values obtained are below the limits given for base materials. This will also be confirmed for HPC as the Project progresses.

5.40. The following applies to the limit on nickel content in the beltline. The RCC-M formula for the prediction of irradiation embrittlement is calculated using base material data, and is conservative for the weld metal, up to a nickel content of 1.2% (hence the upper limit on Ni in welds in the beltline region). For HPC, the Ni content in RPV welds is restricted to a lower value than this.

5.41. The HPC RPV is currently under procurement, with most RPV forgings already made. Welding of these to form the RPV itself is due to start later in 2017.

Description of ageing management programmes for RPVs

5.42. All of the main elements that make up an AMP for HPC are in development, and some are more advanced than others. The elements that are being developed as part of the HPC AMP are as follows:-

- RPV degradation (embrittlement) to be monitored through an Irradiation Surveillance Programme (ISP)

- Assessment for the potential for initiation of defects due to fatigue to be managed through a transient monitoring programme

- In-Service Inspections of the RPV, in particular the welded areas to confirm continuing absence of structurally significant defects

- Monitoring and control of Primary water chemistry to ensure corrosion is minimised

- Operating Experience (OPEX) reviews to understand feedback from other plants and relevant research (e.g. Thermal ageing of ferritic steels)

- A Boric Acid Surveillance Programme to enable early identification of leaks

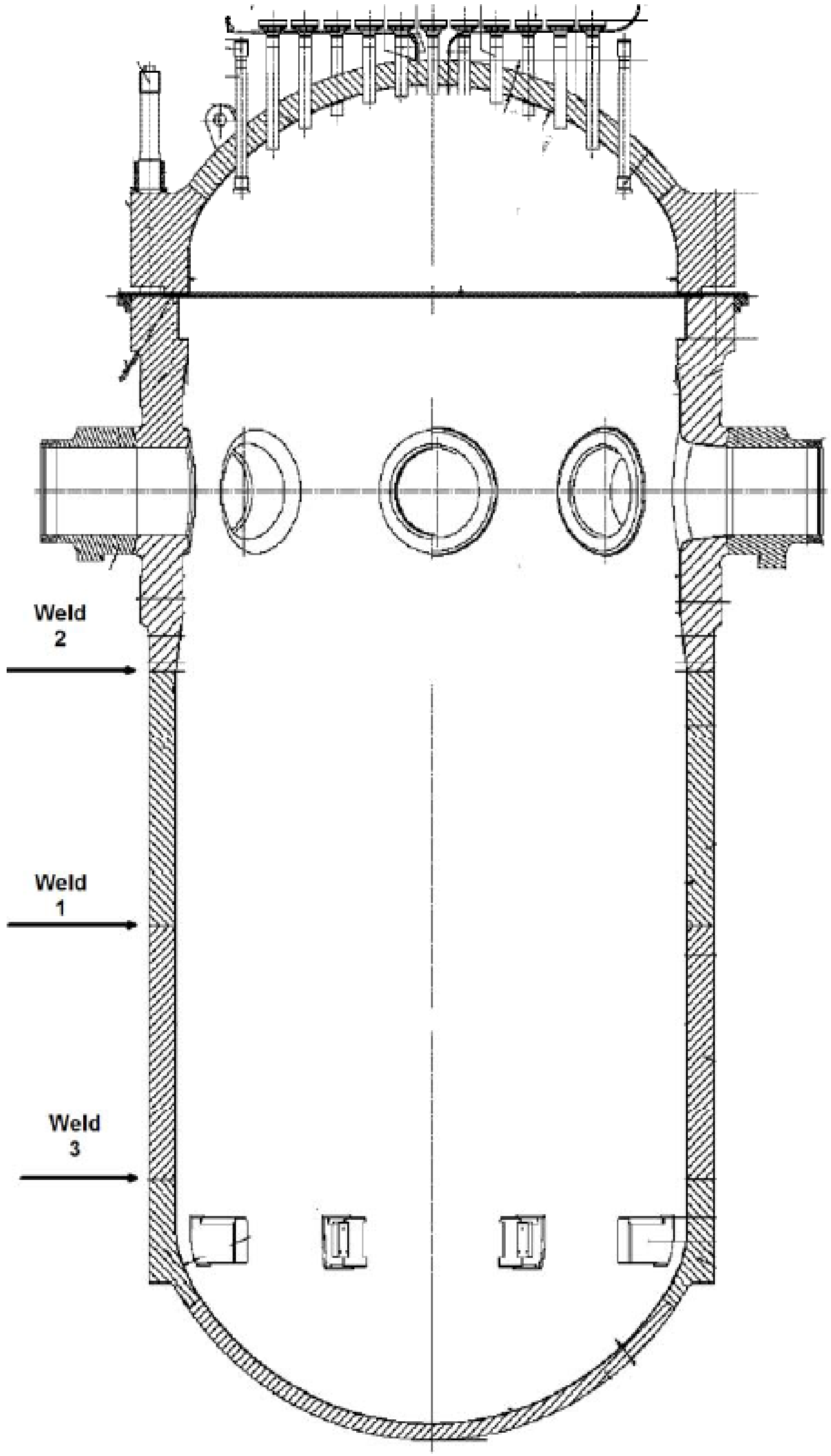


Figure 5.3 – EPR™ RPV

5.43. In addition to the above, there are a number of aspects that are currently being implemented which provide confidence that ageing is being considered in the current design and procurement phases of the HPC Project:-

- Quality Assurance (QA) arrangements specified to contractors for design and procurement activities (and all other quality related activities)
- Rigorous review by NNB (and its Responsible Designer) of Equipment Specifications and lower tier documents, including technical manufacturing programmes and process qualification documents to confirm ageing is appropriately considered
- Additional chemical controls specified on RPV forgings and weld consumables, and reviewed at the end of the material procurement phase to confirm forgings have met the specified requirements

Scope of ageing management for RPVs

5.44. The main components that constitute the pressure boundary are the forgings listed below, and their associated welds:-

- Upper head (with shrink fitted and welded tubes for RCCAs, Instrumentation and other small penetrations)
- Flange ring (with holes for closure studs)
- Nozzle shell with integral flange (with eight penetrations for the forged nozzles to be welded onto)
- Two core shells
- Transition Ring
- Lower Head

5.45. The ageing mechanisms that are considered for all of the above components are thermal ageing and fatigue. Irradiation embrittlement is also relevant, mainly for the two core shells, but also for the transition ring and the Nozzle shell. Irradiation embrittlement is arguably the dominant ageing mechanism for the RPV.

5.46. General corrosion of the ferritic pressure boundary is effectively eliminated due to the use of surface cladding of the above components. The surface cladding of the RPV in stainless steel is subject to sampling to ensure that the requirements relating to composition and delta ferrite are met. A minimum of two layers of stainless steel cladding are applied to the inner surface of the vessel wall to eliminate corrosion. The first layer is grade 309L stainless steel, and the second and following layers needed to meet the required thickness are grade 308L stainless steel. General corrosion of the stainless steel cladding is minimised by the optimisation of the primary water chemistry and the injection of Zinc Acetate into the coolant water. Visual inspections for leaks of the Reactor Coolant System (RCS) will be undertaken following outages to confirm continued integrity of the pressure boundary.

5.47. Stress corrosion cracking (SCC) is also effectively eliminated through the use of SCC resistant stainless steels in the Upper head penetration welds and appropriate control of water chemistry through the injection of Zinc Acetate as described above. It is also worth noting that the elimination of Bottom-Mounted Instrumentation (BMI) penetrations for the EPR™ has also removed a potential zone of elevated SCC risk. The In-Service Inspection of these welds is being considered as part of the development of the ISI programme for HPC.

5.1.2 Ageing assessment of RPVs

5.1.2.1 Operating reactors (EDF-NG)

5.48. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

5.49. The elements that make up Sizewell B's RPV AMP are:

- RPV degradation through the monitoring for such effects as irradiation embrittlement and thermal ageing.
- RPV fatigue management through transient monitoring arrangements.
- In-service inspections of RPV, in particular the welds.
- Equipment qualification arrangements, in particular conservative design using an internationally recognised design code.
- R&D collaboration through industry bodies, universities and research organisations.
- Operating experience (OPEX) reviews.
- Fracture analysis where the safety case postulates start of life defects and using conservative crack growth rates the crack is assumed to grow such that a margin of safety is still present at the end of life. This is applied to all high stress areas of the RPV.
- Quality assurance arrangements during manufacture and fabrication of the RPV.
- A boric acid surveillance programme to give an early warning of failure, through the identification of leaks and boric acid corrosion sites.

All these elements are shown graphically in Figure 5.3.

5.50. Ten yearly periodic safety reviews provide an additional opportunity to review these arrangements. EDF-NG also has in place a number of expert bodies to provide oversight and strategic direction for ageing management activities relating to the RPV.

5.51. *The following paragraphs describe the ageing assessments, the basis on which the programmes were derived and how the outputs are used. Protocols are in place to make* sure the surveillances take place and the RPV remains compliant with the safety case. Collectively it is claimed they provide a comprehensive picture on a continuing basis, of the degradation mechanisms affecting the RPV. The outputs from ageing management programmes are also detailed below for each identified degradation mechanism.

Fatigue management (SP3)

5.52. The RPV has been designed to the requirements of ASME III for a plant lifetime of 40 years. The fluid system transient events which were important enough to be considered in the design of the RPV were based on the American National Standards Institute (ANSI) developed criteria defined in ANSI N18.2. These broadly correspond to the ASME operating conditions. These design transients are specified in transient schedules which provide a conservative definition, in terms of both severity and number of occurrences.

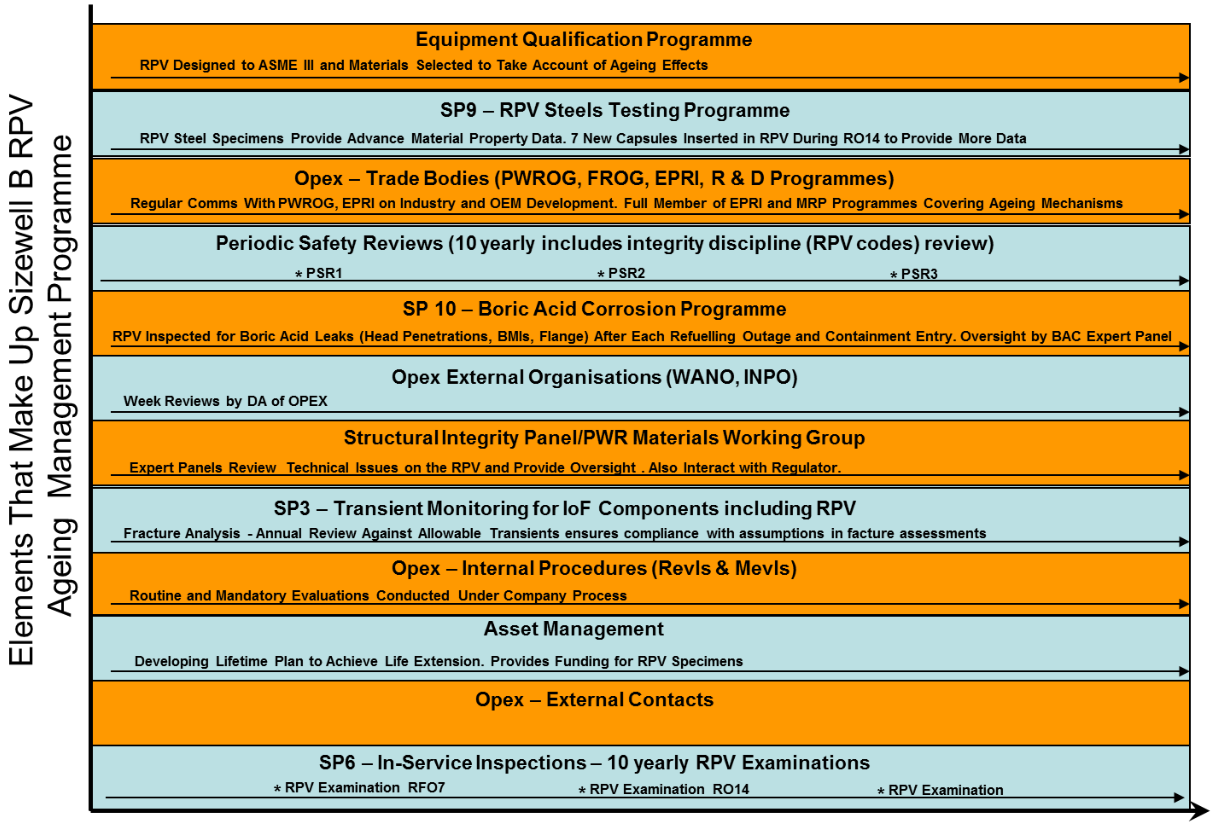


Figure 5.3 - Timeline – Sizewell B Design Life (40 years)

5.53. The ANSI criteria established four separate categories of plant conditions that were considered in the design. These ‘Conditions for Design’ followed a basic principle that the most frequent occurrences result in little or no adverse consequences and that the extreme transients have the potential for the greatest adverse consequences at the lowest probability of occurrence. The ‘Conditions for Design’ are very similar to the ASME ‘Operating Conditions’. These are used to assign the correct operating conditions to the appropriate service levels and hence stress limits applied to the design of RPV components, thereby giving assurance that integrity was not comprised by operating transients.

5.54. The Sizewell B design basis transients are based on the Westinghouse design schedule of transients and are applicable to the RPV and other RCS components. The method of presentation provides details of variations in load, fluid pressure, temperature and flow given in the form of design transients together with an estimate of their frequency of occurrences. Design transients are essentially bounding transients intended to cover all the operational, test and fault events Sizewell B may be subjected to. The objective when defining design transients is to strike a balance between keeping the number of different transients to a minimum to simplify the equipment fatigue analysis against introducing undue conservatisms by covering too large a group of events with the same design transient.

5.55. The Sizewell B RPV is assigned as an IoF component and in addition to the ASME code design additional fracture assessments have been undertaken. The transients used in this additional assessment are largely the same as those used in the ASME assessment, however for the RPV components fracture analysis transients specified in a separate transient schedule are used. They ensure the analysis remains conservative without providing unnecessary constraints. These transients are monitored under the surveillance programme (SP3).

5.56. The operational transients are reviewed and recorded within two months of the occurrence as required by the programme. The cumulative number of transients are recorded and compared with the allowable number of occurrences. The rate of occurrence is extrapolated to predict whether the allowable number of occurrences is likely to exceed the 40 year component lifetime. In the event that the allowable number of occurrences is likely to be exceeded senior staff are notified and the issue is recorded with company processes. The cause is investigated and necessary actions taken. There is also an annual review by the corporate Design Authority function.

5.57. In order to ensure this surveillance is appropriately enacted there are two surveillance test procedures in place. One test procedure administers a monthly review of transient data, while the other the annual review of transients. These protocols are part of the company Work Management process, with the surveillances undertaken by the nominated programme owner.

5.58. Oversight of transient monitoring is provided by a SIP under which is an EDF-NG Fatigue Management Working Group.

In-Service Inspections (SP6)

5.59. Surveillance Programme 6 (SP6) details the in-service inspection requirements, which for the RPV are in general accordance with the scope and frequency of ASME XI. This forms an important part of the forewarning of failure for the RPV safety case. The original basis for the in-service inspections during the First Inspection Interval was based on the 1989 version of ASME XI. The current Inspection Interval uses the 2007 version of the ASME XI code. The code changes are implemented at Sizewell B as a result of the ten-yearly update requirements in the programme and are managed through the safety case modifications process. The programme schedules set out the frequency and examination requirements by component. In the case of the RPV the focus of attention is the welds. As the RPV has an Incredibility of Failure status, the inspection criteria for welds is based on those given in the safety case thereby linking inspections to the postulated defects calculated for end of life.

5.60. EDF-NG pay particular attention to the developments in the ASME XI code and these are periodically reviewed under normal processes and at the end of each Inspection Interval.

5.61. In following the requirements of ASME XI an inspection plan is developed well before each refuelling outage. This includes the requirements for the RPV examinations. In order to maintain compliance with the safety case a surveillance test procedure is in place to confirm the inspections have been approximately executed. There are two further surveillance test procedures, one covering the surveillance requirements to monitor the OPEX of SCC of RPV components, the other covers ASME code requirements for thermal fatigue monitoring to ensure the inspection programme remain adequate.

5.62. Oversight of in-service inspections falls under the auspices of the SIP. The SIP meets frequently during inspection campaigns to consider any results that fall outside the acceptance criterion.

RPV steel programme (SP9)

5.63. The surveillance programme monitors any changes in the mechanical properties of the RPV steel material throughout its life. SP9 comprises two elements covering the effects of irradiation and thermal ageing. The effect of strain ageing in combination with irradiation and of strain ageing in combination with thermal ageing is also examined by incorporating pre-strained samples in both elements.

5.64. When Sizewell B began operation, eight specimen capsules were positioned in guide baskets attached to the outside of the neutron shield pads and positioned directly opposite the centre portion of the core. All eight capsules have now been removed and specimens tested. Specimens are irradiated in the ratio approximately 3:1 based on the proximity to the core in relation to that of the RPV Core Shell. The recent capsule removed during RO14, has an equivalent of 60 years operation, while Sizewell B is currently 22 years old. The programme ensures that the RPV steel remains within the limits assumed in the definition of the Reactor Coolant System pressure and temperature (P-T) limits and used in the fracture analysis.

5.65. The evaluation of neutron damage is based on pre-irradiation testing of Charpy V-notch, tensile and compact tension specimens and post-irradiation testing of Charpy V-notch, tensile and compact tension fracture mechanics test specimens. The specimens are taken from actual parent materials and representative welds as the RPV.

5.66. The programme is directed towards evaluation of the effect of radiation on the fracture toughness and ductility of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach. The programme is based upon the American Society for Testing and Materials (ASTM) standard ASTM E-185. Included in the specimen capsules are dosimeters. The dosimeters permit evaluation of the flux experienced by the specimens and the vessel wall indirectly. In addition, thermal monitors made of low melting point eutectic alloys are included to monitor the maximum temperature of the specimens.

5.67. In the context of this programme, the relevant limits are those related to the shift due to ageing in the ductile-brittle transition temperature and to the actual transition temperature or ‘Nil-Ductility Temperature’, at the end of life of the RPV.

5.68. The Action Conditions within the surveillance programme include for any increase in the Nil-Ductility Temperature as defined in ASME III being out of limits for the irradiated RPV Core Shell and associated welds. Appropriate actions and completion times are stipulated and documentation (System Test Procedures) signed off to ensure compliance with the safety case for the vessel.

5.69. The analysis carried out to derive the RCS pressure and temperature limits and the RPV fracture analyses have used different methodologies (i.e. ASME III Appendix G and ‘R6’ - an EDF-NG generated structural integrity code respectively). The ductile-brittle transition temperature has been predicted by two different routes to be consistent with those overall methodologies.

5.70. A percentage of the specimens have a 5% pre-strain applied to model strain ageing. It is believed no other utility does this. A further seven capsules have been inserted in the RPV during RO14 to provide more data points for life extension purposes.

5.71. SP9 also contains a thermal ageing element. The test specimens are also the same as the RPV steel and commenced their ageing treatments in advance of the vessel entering service. They are aged at the highest normal operating temperature. There is no attempt to accelerate thermal ageing effects by treatment at higher temperatures. The results are analysed and compared with the relevant ductile-brittle transition temperature and if necessary the relevant Nil-Ductility Temperature limit.

5.72. To maintain compliance with the RPV safety case a surveillance test procedure ensures the review of results of test specimens is appropriately reviewed and fed back to ensure the operational limits are not contravened.

5.73. An EDF-NG expert panel called the PWR Materials Group provides the necessary oversight on SP9 matters.

Boric acid corrosion programme (SP10)

5.74. The programme provides a multi-layered approach to identifying, minimising and controlling leakage of borated water from the primary circuit including the RPV. Although there are Technical Specification limits for boric acid leakage it is known that smaller leakages can still give rise to corrosion of ferritic steel components. The programme also searches for boric acid corrosion sites.

5.75. In the case of the RPV visual inspections for leaks take place each refuelling outage at normal operating pressure/normal operating temperature prior to return to service. The areas include penetrations and the RPV Closure Head Seal. The Head Seal, comprising two nickel alloy rings has monitoring of the interspace. The system is regularly monitored for fluid and tested as appropriate for radionuclides (Note the head seal monitoring is not part of the programme, it is part of the containment leakage monitoring arrangements).

5.76. The programme sets out the appropriate actions following the discovery of a boric acid leak and an EDF-NG Boric Acid Corrosion (BAC) expert panel provides oversight to the programme. There is regular leakage monitoring of RCS leakage into containment and the BAC programme owner issues monthly reports to the panel members.

5.77. The programme is based on industry experience, most notably Davis Besse, and guidance used in the United States from the US Nuclear Regulatory Commission (NRC Generic Letter GL 88-05) and Westinghouse (WCAP-15988-NP). Compliance routines are administered via a surveillance test procedure.

Applicable ageing management standards

5.78. An adequate ageing management programme (AMPs) for a nuclear power plant is fundamental to ensure the long term integrity of the RPV. This was recognised early and provisions were made in the original design and manufacture. The Sizewell B safety case contains all the elements outlined above and form part of the site licence conditions.

5.79. In 2009 EDF-NG undertook a wide ranging review of the adequacy of Sizewell B activities in managing material degradation issues. The review concluded that degradation mechanisms were being proactively managed for the RPV.

5.80. Over time there have been a number of ageing management initiatives developed across the nuclear industry, such as the Generic Ageing Lessons Learnt programme under the US NRC. Also the Nuclear Engineering Institute (NEI) has done work to ensure the correct management controls are in place and is documented under NEI-03-08.

5.81. The theme of ageing management has more recently been progressed through the International Atomic Energy Agency (IAEA) who has developed the International Generic Ageing Lessons Learnt (IGALL) programme. The programme seeks to develop and maintain documents and a database to provide a technical basis and practical guidance on managing ageing of components and structures on nuclear power plants. It is appropriate to compare the provisions at Sizewell B with those of a modern ageing programme, therefore, the remainder of this section focuses on how Sizewell B compares to the provisions laid out in the IGALL.

5.82. The applicable RPV AMPs in the IGALL programme are considered to be:

- **AMP101 - Fatigue Monitoring.** This programme outlines provisions for monitoring and tracking those transients for the related location to ensure cumulative fatigue damage remains within allowable limits.
- **AMP102 – In-Service Inspections.** This includes the requirements for the periodic visual, surface and volumetric inspections and leakage tests for pressure retaining components. How components are evaluated and monitoring of ageing effects.
- **AMP103 – Water Chemistry.** The programmes aim is to mitigate loss of material due to corrosion. For PWRs it refers to EPRI PWR Primary Water Chemistry Guidelines.
- **AMP104 – Reactor Head Closure Stud Bolting.** This AMP covers in-service inspections and preventative measures to detect and manage cracking and loss of material of the closure bolting components of the RPV Head.
- **AMP110 – PWR Boric Acid Corrosion.** This monitors the condition of the reactor coolant pressure boundary, and therefore the RPV, for sources of borated water. The programme includes provisions for initiating evaluations and assessments when leakage is discovered.
- **AMP111 – Cracking of Nickel-Alloy Reactor Coolant Pressure Boundary Components.** The programme addresses the issue of nickel-alloy components and consequential loss of material due to boric acid induced corrosion. The programme recognises the long term inspection requirements to mitigate the threat from stress corrosion cracking.
- **AMP118- Reactor Vessel Surveillance.** The programme focuses on the RPV steels exposure to the high energy neutron flux which results in irradiation embrittlement. Its main aim is to establish the principles for RPV surveillance programmes which in general are to provide sufficient material data and dosimetry to monitor embrittlement until the end of the RPV design life and to determine any requirements for restriction on operating parameters e.g. pressure-temperature limits.

5.83. AMP 118 outlines a programme based on the measurement of increases in Charpy V-notch transition temperature at an established reference level of impact energy and the drop in the upper shelf energy as a function of neutron fluence and irradiation temperature. The IGALL programme provides the nuclear industry with the most up to date arrangements agreed by Regulators and plant operators.

5.84. In comparing the AMPs to Sizewell B provisions, in many cases there is a direct relationship, and it is concluded that the Sizewell B surveillance programmes all meet the intents of the AMPs under IGALL and in the case of in-service inspections and RPV steels monitoring the Sizewell B arrangements exceed those of the respective AMPs.

Research and Development

5.85. EDF-NG is involved with a number of R&D activities. In terms of industry bodies EDF-NG is a member of the PWROG and EPRI. Both organisations undertake research for the PWR community. EDF-NG provides funding and collaborates in projects and has access to the outcomes of the work.

5.86. EDF-NG is also a member of the Materials Ageing Institute and regularly attends functions organised by that body. Membership also provides access to published material such as Materials Ageing in Light Water Reactors.

5.87. Access to materials testing and developments in structural analysis are through the internal development of 'R Codes' and university collaborations. The company has signed agreements for collaboration with universities of Manchester, Bristol, Strathclyde, at Imperial and the Open University. Arrangements are also in place with The Welding Institute covering developments in inspections techniques.

5.88. EDF-NG has a company process to support the sustainability of plant operations through research related to maintaining current and developing future technical capability. It describes the management of the R&D programme and the principles with which a licensee's nuclear safety related research arrangements should comply (paragraph 2.80).

5.89. Given the involvement with the different bodies involved with R&D it may be concluded that EDF-NG has reasonable access to R&D activities related to the RPV.

Operating experience (OPEX)

5.90. OPEX forms a key dimension of ageing management for Sizewell B, This is due to the age of the Sizewell B, which was built later than a number of sister plants such as Wolf Creek and Callaway power plants. The design has benefited from improvements due to lessons learnt from earlier PWRs. In the case of the RPV this includes the avoidance of a beltline butt weld in the highly irradiated zone of the RPV Core Shell.

5.91. Great emphasis is placed on retrieving worldwide OPEX to ensure any countermeasures can be taken at the earliest opportunity. An example of this being improvements in the way the Bottom Mounted Instrument (BMI) penetrations are examined.

5.92. EDF-NG has established a number of protocols to formalise the review of OPEX. This includes mandatory or routine reviews from industry bodies such as the PWROG (when they issue Nuclear Safety Advisory Letters (NSALs) or Technical Bulletins) and WANO notices. Company procedures provide the impetus to undertake the reviews. The OPEX arrangements were reviewed at the last PSR and found to be appropriate. Since the last PSR the OPEX arrangements have been consolidated into the arrangements for Organisational Learning.

5.93. Part of the formal training requirements in EDF-NG is covered under Mentor Guides for specific positions. The relevant Mentor Guide for RPV integrity includes an appreciation of OPEX, what it is, what the sources are, its relevance to safe operations and who are the right organisations or people to go to for advice. Emphasis is placed on OPEX during the training. EDF-NG has regular two way dialogue with other PWR operators and industry bodies such as EPRI. There is also an operational liaison group within EDF-NG with prime responsibility for interfacing with the rest of the EDF organisation.

5.1.2.2 Hinkley Point C (NNB)

5.94. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

5.95. The RPV has been designed and constructed to the requirements of the RCC-M Code for a lifetime of 60 years. The transient (normal, upset, emergency and faulted) conditions taken into account in the design of the RPV are bounding of the conditions occurring during operation of the plant. The transients selected constitute a reference base for the evaluation of the RCS in order to ensure the integrity of the components.

5.96. The mechanical design rules formulated in the RCC-M and the stress limits specified in Level 1 of the RCC-M are intended to prevent the following types of damage to the vessel:

- Excessive deformation and plastic instability,
- Progressive deformation and fatigue cracking,
- Fracture

5.97. The approach taken towards the prevention of progressive deformation and fatigue-induced initiation of defects in the RPV is based on experience that has been gained through the operation of conservative and known designs, as well as worldwide operating experience. The design assumes a conservative number of pressure and thermal transients to take account of through-life operations.

5.98. As stated previously, Irradiation embrittlement is the dominant ageing mechanism for the RPV. The RPV core shell design is based on the dimensions of the core. The diameter of the core shells is designed to fulfil the design criterion for the core shell and the core shell weld such that the Reference Transition Temperature (nil-ductility) End of Life (RT_{NDT} EOL) will be lower than 30°C. With the current diameter of core shell, an end-of-life fluence of around 1.26×10^{19} n/cm^2 (E > 1 MeV) is reached assuming certain operating conditions. This is much lower than that of current PWRs due to the presence of a heavy reflector located inside the core barrel, which surrounds the core and provides neutron protection for the RPV wall.

5.99. The design intent is achieved by verification that the EOL transition temperature RT_{NDT} remains below 30°C. In all French RPVs, weld metal EOL RT_{NDT} is bounded by the properties of the base metal due to tighter chemical composition controls on weld consumables. This situation is mainly due to the low level of the initial RT_{NDT} of weld metal compared to that of base metal. The situation is expected to be the same for HPC, however, the weld metal is also included in the ISP.

5.100. The result of neutron embrittlement of the RPV material is an increase in its yield strength and in its ductile to brittle transition temperature, which thus entails an increased risk of fracture. The factors determining the amount of irradiation embrittlement which the RPV material experiences have been well established for the low-alloy steel grades used for RPV construction and are covered by the RCC-M code. The expected fluence for the EPR^{TM} project is far below the limit of validity of the formula used to determine irradiation embrittlement: 8×10^{19} neutrons / cm^2 (E > 1 MeV), even in the case of life extension to 80 years. Another methodology to analyse the RPV irradiation effects, based on displacements per atom (dpa), will also be applied.

5.101. During the RPV design stage, an analysis of the risk of fracture of the RPV is performed using material properties integrating ageing through formulas predicting irradiation embrittlement. In addition, the actual embrittlement of the materials is measured periodically during service; mechanical tests are carried out on test specimens inserted inside the RPV, and which are subjected to a greater neutron flux than the vessel wall. The test specimens provide information on material ageing at an early stage, allowing the specified operating conditions to be updated if necessary.

5.102. Although the transition curve between brittle and ductile behaviour is relatively steep, the evolution of the transition temperature RT_{NDT} with time (i.e. with neutron fluence) is progressive. It is considered in RCC-M (ZG.6122) that the average irradiation effect is proportional to the square root of the neutron fluence. The latest formulas expressed by EDF, based on results of surveillance programmes, give an envelope exponent. The resulting variation with time of the shift in transition temperature should be very close in the two cases.

5.103. This means, for example, that an extension of the design life from 60 to 80 years would result in an increase of the transition shift by 15% of the calculated shift. The total shift remains low compared to international experience obtained on existing PWR plants.

5.104. This, combined with the safety margins imposed at the design stage and the fact that the surveillance capsules will provide information on actual material ageing sufficiently in advance, guarantees that there are no unanticipated cliff edge effects which could impair vessel safety.

5.105. Thermal ageing is not a major concern for RPV materials, considering the operating temperatures and low contents concerning residuals that could contribute to thermal ageing sensitiveness (phosphorous, antimony, tin and arsenic). Any eventual change in properties due to thermal ageing would be bounded by the properties of the Pressuriser which operates at higher temperature, and has been studied extensively previously.

5.1.3 Monitoring, testing, sampling and inspection activities for RPVs

5.1.3.1 Operating reactors (EDF-NG)

5.106. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

Monitoring and testing

5.107. The RPV is continuously monitored for flow rate, temperature and pressure variations under Surveillance Programme 3. Transients considered important in the design process are based on the American National Standards Institute (ANSI) N18.2 – 1973 for PWRs. These design transients are specified in transient schedules which provide a conservative definition in terms of severity and number of occurrences, which the RPV is expected or postulated to experience over the plant lifetime.

5.108. The ANSI criteria established four separate categories of plant condition to be considered and are very similar to the ASME operating conditions. The plant conditions for design are:

Condition 1 - Normal Operation (Normal under ASME)

Condition 2 – Incidents of Moderate Frequency (Upset under ASME)

Condition 3 - Infrequent Incidents (Emergency under ASME)

Condition 4 - Limiting Faults (Faulted under ASME)

5.109. In accordance with the requirements of the ASME code conditions 1, 2 and Test design transients are used for monitoring fatigue damage of the RPV. Transients monitoring is the responsibility of the SP3 Programme Owner who logs the transients within two months of the occurrence. An annual report is compiled identifying the number of occurrences against the cumulative and design allowances. The review is assessed by the Design Authority. To date Sizewell B RPV has operated well within the allowable transients for Conditions 1 and 2.

5.110. By their very nature Conditions 3 and 4 transients are such that should they occur they would be subject to an extensive engineering evaluation or in the extreme repairs to structural components. This would then take account of the peak stresses and would form part of any justifications to demonstrate integrity and safe return to service.

5.111. During construction the Sizewell B RPV was subjected to two hydro-static tests (21.5MPa) at 1.25 times the design pressure (17.2 MPa) and no further such tests have been carried out or are envisaged.

Sampling

5.112. Sizewell B has a comprehensive RPV material sampling surveillance programme (SP9). The objective of the programme is to confirm through the life of the plant that the properties of the RPV steel remains within the limits assumed in the safety case that relates to pressure and temperature and the RPV fracture analysis.

5.113. The main focus of SP9 relate to the shift due to ageing of the ductile-brittle transition temperature and the actual transition temperature at the end of life of the RPV. The programme is expected to confirm that the shift does not exceed temperature limits for the RPV Core Shell forging (at quarter wall thickness) such that pressure-temperature limits remain valid and confirm the shift does not exceed temperature limits for the irradiated shell region (forging and weld).

5.114. The programme consists of archive RPV material specimens (SA 508 Class 3 ferritic steel) which were placed in eight capsules close to the reactor core at start of operation and hence were subjected to a higher dose rate than the RPV Core Shell. The lead factor is of the order of 3 to 1. Each of the 8 capsules contains approximately 120 specimens made up of parent material and weld samples. Around 50% of the specimens have been subjected to a 5% pre-strain to investigate strain ageing effects. It is believed that Sizewell B may be the only world-wide PWR that imposes a 5% pre-strain on materials in its steels surveillance programme.

5.115. The irradiation surveillance capsules were removed periodically and have been tested in a test laboratory capable of handling active samples. Suitable specimens enable Charpy Tests, Compact Tension Tests (fracture toughness tests) and tensile tests to be performed. Results to date remain within limits, however it was recognised that more data was needed towards the end of life particularly as life extension is currently proposed. A further seven capsules were inserted in RO14 (April –June 2016) in order to provide more data points close to end of life. Approximately 50% of the specimens have 5% pre-strain. The future capsule removal programme is set down in the surveillance programme.

5.116. In addition Sizewell B has a thermal ageing programme which runs approximately two years in advance of operation. Samples are aged out of circuit in a long term ageing furnace. Protection from oxidation is by use of a controlled furnace atmosphere or by encapsulation.

Inspection activities

5.117. The Sizewell B RPV underwent significant diverse and redundant inspection prior to service. In accordance with the in-service requirements of ASME XI the RPV has been subjected to two ten-yearly inspections since the beginning of commercial operation in 1995. RPV inspections are conducted using a combination of visual and volumetric examinations. The volumetrics are conducted using an automatic ultrasonic (UT) examination technique through the use of a robotic device housing the UT probes, see Figure 5.4. The examinations are subjected to a validation process which includes blind trials whereby the inspection procedures, equipment operators and the UT probes are tasked with finding known defects in large archive test block forgings to demonstrate defects of a given size may be detected. The validation process is managed by a company independent to EDF-NG known as the Inspection Validation Centre. The actual locations and orientation of defects are only known to the Inspection Validation Centre. These arrangements exceed those of ASME XI.

5.118. The sizes of defect that are tested for under the blind trials and in-service are consistent with those postulated in the safety case and used in the fracture assessment. These form the acceptance criteria for the RPV examinations. In reality the reporting of defects is based on a reporting threshold, which is much smaller than that used in the safety case. This supports the forewarning of failure leg of the safety case and provides an additional layer of conservatism.

5.119. Because EDF-NG hopes to operate Sizewell B beyond the original forty year design life the opportunity was also taken to examine the whole of the RPV Core Shell in RO14. This is not a requirement of the ASME XI code. It also represented an opportunity to verify that there were no hydrogen flakes present in the RPV Core Shell similar to those found at Doel 3. This examination exceeded the WENRA recommendations on the matter.

5.120. In accordance with other PWRs the nozzle inner radii are inspected via an enhanced visual. Ultrasonic Testing (UT) is no longer performed. This was justified previously using the company safety case modifications process and is now embedded in the RPV safety case.

5.121. The robotic arm is also used to inspect the nozzle to safe end welds on all eight RPV penetrations.

5.122. In RO14 there was a change to the way in which the Bottom Mounted Instrument penetrations (BMIs) were examined based on OPEX from other PWRs. The penetrations use Alloy 600 parent material and Alloy 82/182 weld material, both are susceptible to stress corrosion cracking which has been observed on some PWRs worldwide. Sizewell B accordingly deployed UT and Eddy Current Testing for the first time on each penetration. No significant defect indications were found.

5.123. During RPV examinations provision is made to follow up any unexpected degradation. However, no such degradation has been found and in all the RPV examinations no defects of concern have been found.

5.124. As previously stated Sizewell B replaced its original RPV Head in 2006. The RPV Head was therefore not included in the scope of examination in RO14. These will take place in RO15 and will follow the requirements of ASME XI code and includes the main flange to dome weld (P-R weld) and the Control Rod Drive Mechanism (CRDM) penetrations. The Head entered service with no defects present following comprehensive examinations.

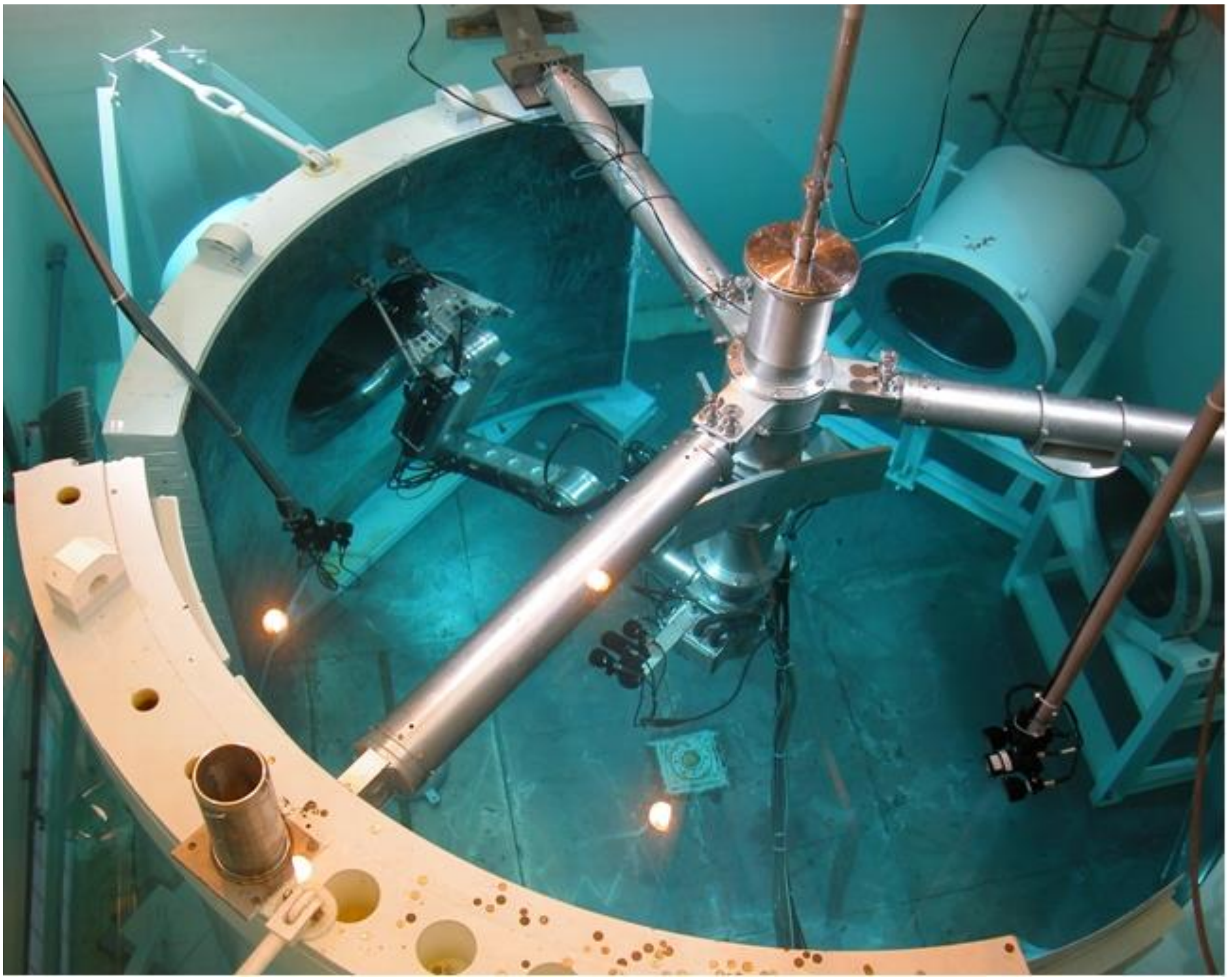


Figure 5.4 The robotic arm during validation (blind) trials in RFO14

5.1.3.2 Hinkley Point C (NNB)

5.125. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

5.126. The monitoring and testing programmes that may be proposed for the HPC RPV are described above in previous sections. More information is given below on the Irradiation Surveillance Programme (ISP).

5.127. In the ISP, the evaluation of radiation damage is based on testing of a combination of pre-irradiation and post-irradiation Charpy V-notch, tensile and 1/2 T (thickness) Compact Tension (CTJ) specimens. The programme is directed at evaluating the effect of irradiation on the mechanical properties of reactor vessel steels during service.

5.128. The vessel monitoring programme uses specimen capsules housed in holders attached to the outside of the internal vessel barrel, and positioned directly opposite the central section of the core. These capsules can be removed when the vessel closure head and the upper core support structure are removed. All capsules contain specimens of the RPV base metal of the two core shells of the reactor, of the core weld metal and of the associated heat-affected zone metal. Each capsule encloses tensile test specimens, Charpy V-notch specimens (which contain base metal, weld metal and metal from the heat-affected zone) and Compact Tension specimens. Archive materials are kept in sufficient quantities for additional capsules.

5.129. The exposure of the specimens to neutrons occurs at a faster rate than that experienced by the vessel wall, as the specimens are located between the core and the vessel. Since these specimens experience accelerated irradiation and are actual samples from the materials used in the vessel, the transition temperature shift measurements between non-irradiated and irradiated specimens are representative of the vessel later in life.

5.130. Correlations are established between the calculations of the fluence and the measurements of the irradiated samples in the surveillance capsules. The surveillance capsules are located within the reactor vessel so that the specimen irradiation history duplicates as far as possible, within the physical constraints of the system, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel.

5.131. Due to the presence of the heavy reflector, the neutron energy spectrum is not strictly equivalent between the locations of the surveillance specimens and the RPV wall. The degree to which the heavy reflector and the specimen environment will induce a difference in neutron flux and energy spectrum between the two locations will be considered in the analysis of the surveillance specimen data.

5.132. The schedule for the removal of the capsules for post-irradiation testing will account for the spectrum effect by considering the dpa as well as the fluence dose parameter. The detailed schedule is being developed as part of the ISP definition.

5.133. Start-up and shutdown operating limits are based on the specified properties of the core region materials of the RPV. The design operating curves are calculated assuming a period of reactor operation such that the beltline material will be limiting. Beltline material properties degrade with irradiation exposure, and this degradation is determined in terms of the adjusted reference nil-ductility temperature, which includes a reference nil-ductility temperature shift ΔRT_{NDT}.

5.134. Predicted ΔRT_{NDT} values are derived using two parameters, these being the effect of flux and the effect of copper and phosphorus contents on the shift of RT_{NDT} for the materials located in the beltline region (base material and weld material).

5.135. Pressure / temperature curves are calculated according to the RCC-M appendix ZG methodology, with a conventional quarter thickness defect located on the inner diameter of the core shells. The maximum fluence is determined for various selected time periods through the reactor life and the analysis is performed at the deepest point of the defect which covers the crack front intersection with the surface of the vessel wall.

5.136. A comparison between the operating domain and a Pressure-Temperature curve derived from a heat-up from 15°C to 303°C (with a gradient of 40°C/h) transient shows that:

- the P-T curve derived from fracture analysis does not result in a limitation for the operating domain, which arises from functional and fatigue aspects,
- there can be no interference between them even by making the P-T curve more pessimistic by up to 30°C.

5.1.4 Preventive and remedial actions for RPVs

5.1.4.1 Operating reactors (EDF-NG)

5.137. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

5.138. The following outlines the key preventative and remedial actions under the various surveillance programmes and the RPV components most affected

Surveillance Programme 3

5.139. Should the RPV transient count get close to the annual allowable or the projected life time allowable actions conditions will be applied. The actions identified in the programme range from minor reviews to full engineering studies, which demonstrate acceptability of increasing the lifetime allowable occurrences of the particular design transients. Depending on the severity of the transients under consideration varying timeframes are ascribed for completion and in an extreme case lead to operational constraints.

5.140. The first step will typically require an investigation into the cause of any increase in the rate of occurrences and to determine whether the transients may be reallocated. This may be followed by reviewing the necessity to increase the frequency of surveillances.

5.141. The surveillance programme allows for further engineering studies to take place before extreme operational restraints are applied.

5.142. The surveillance programme considers the RPV as a whole structure and does not break the transients down to a component level. High stress locations in the RPV are assessed with the bounding locations covered in the RPV demonstration of integrity safety case and use the design schedule of transients for IoF components in the analysis.

Surveillance Programme 6

5.143. The in-service inspection programme provides a comprehensive forewarning of failure element to RPV integrity.

Surveillance Programme 9

5.144. The surveillance programme has a built in lead factor that allows the feedback from test results to be fed back into the assessment that determines the pressure-temperature limits, in a timely manner. Any changes to P-T limits can be used to change the way the RPV is operated. The inclusion of a thermal ageing element ensures that any affects from thermal damage are monitored and the RPV safety case limits remains valid.

5.145. The surveillance programme is only applicable to the RPV Core Shell and the upper and lower Core Shell welds as the accrued dose levels bound the other RPV components.

Surveillance Programme 10

5.146. The Boric Acid Surveillance Programme was directly developed in response to industry concerns over reactor coolant system leakages. The programme generally supplements the in-service inspection programme and typically covers RCS leaks prior to return to service after a refuelling outage. Key locations such as RPV lower penetrations are inspected to ASME XI VT-2 standard to determine the extent of any boric acid leakage with the surveillance conducted at Normal Operating Pressure/Normal Operating Temperature.

5.1.4.2 Hinkley Point C (NNB)

5.147. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

5.148. The preventative actions for controlling and minimising ageing effects will be developed as part of the process of translating safety case requirements into operational technical specification limits. For example, limits will be placed on the rate of change of temperature in heat-up operations, and on hydrogen concentration in the primary circuit water.

5.149. As part of the fatigue transient monitoring programme, operational transients will be reviewed frequently and compared against the design allowable numbers. Remedial actions can be taken if necessary if the end-of-life predicted number is higher than the design allowable. This could include a more accurate calculation of fatigue damage for specific components, or changes to operational sequences to minimise future transient numbers or the severity of transients.

5.150. The ISP results from the Charpy V-notch specimens will be reviewed when capsules are removed and compared to the end-of-life limit on RT_{NDT}. If the results are outside of expectations, then further testing can be done on the CTJ specimens to confirm any changes in actual fracture toughness.

5.151. Results from the ISI programme will be sentenced using appropriate criteria which have yet to be developed. A safety case submission will be produced for returning plant to service following an outage.

5.2 Licensees' experience of the application of AMPs for RPVs

5.2.1 Operating reactors (EDF-NG)

5.152. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

5.153. This section describes whether experience has shown that ageing mechanisms have progressed in line with expectations. It should be noted that under the Sizewell B safety case the fracture analysis for RPV components is based on an end of life analysis for postulated defects at high stress locations. This assertion runs through several of the surveillance programmes, linking them together. For example the start of life defect size is linked to the defect the in-service inspection searches for.

5.154. Sizewell B has now been operating for over 20 years. The current position with respect to the progress of known ageing mechanisms covered by the surveillance programmes is as follows:

SP3 current position

5.155. The latest annual review of transients is showing significant margin of the allowable transients when set against what has actually been seen by the RPV. This demonstrates the conservative nature of the design schedule of transients. It is likely the accrued transient will provide the opportunity to demonstrate integrity for any life extension proposals.

5.156. There are no actual fatigue usage values for RPV components and a strategic review of fatigue management is currently underway.

SP6 latest inspection results

5.157. In 2016 the Sizewell B RPV was comprehensively examined (excluding the RPV Closure Head which will be examined in 2017). The results were extremely positive with no defects of significance reported. Prior to the examination a review of life time (manufacturing) records was carried out to establish the location of minor manufacturing discontinuities. No changes to these discontinuities were found.

5.158. The 2016 (RO14) examinations also included a volumetric examination of the whole RPV Core Shell forging. This was aimed at providing confidence in life extension aspirations but was also used to finally confirm no hydrogen flakes were present (following the type found at Doel 3 and Tihange in Belgium). All the RPV examinations used equipment and personnel that were validated based on 'blind' trials. The whole of the RPV was visually inspected including the RPV Inlet Nozzles to an enhanced visual level.

5.159. The examinations of the BMI penetrations were also satisfactory and now provide a baseline for future examinations.

SP9 results from surveillance capsules

5.160. The last of the eight original RPV steel surveillance capsules was removed in 2016. The capsule had seen approximately 60 effective full power years of operation. The results have yet to be finalised, however early indications are that they are in-line with expectations and do not undermine the predictions for the RPV due to irradiation embrittlement.

5.161. To support longer term operation further capsules (7) were inserted into the RPV at the same refuelling outage. This is intended to provide more data points towards the end of the original design life and beyond.

5.162. Likewise, the results from the thermal ageing programme have generally reported in line with expectations.

5.163. Historically in 2001 some variations in mechanical properties were found following a review and a safety case was produced to justify a small reduction in the properties. No further changes have been required to the RPV safety case.

SP10 Boric Acid leakages

5.164. Inspections to ASME XI VT-2 standard continue. The BAC Expert Panel meets once per cycle or if a leak occurs. The Expert Panel recently met to consider excess liquid in the RPV Head Seal leak detection equipment but this was found not to be from the Reactor Coolant System.

5.165. Routine monitoring of systems inside the containment building have not revealed any leakages.

RPV – Creusot Loire forgings review

5.166. Since 2015 concerns have been raised over the quality of forgings produced at the Creusot-Forge works calling into question the quality of build on some reactor components. The main concern relates to accuracy of manufacturing records and steel making process where the formation of carbon concentrations due to segregation in thick walled forgings has been found on some forgings (but not at Sizewell B). Figure 5.1 shows that most of the RPV forgings were made at Creusot Loire.

5.167. EDF-NG has undertaken a comprehensive review of the RPV and demonstrated the continuing integrity of the RPV based on a number of key factors. These are summarised as:

- The manufacturing process used for the top and bottom domes was unlikely to generate carbon segregation.
- The size of ingot was smaller than those forgings known to have carbon segregation.
- The lifetime records held at Creusot Loire works were consistent (with one small discrepancy) with those held by EDF-NG.
- The ferritic forgings were within specification for levels of carbon (<0.2%).
- The mechanical testing of forging off-cuts was in line with specification.
- There were high levels of independent scrutiny and oversight during manufacture.
- There were comprehensive diverse and redundant inspections during manufacture and pre-service.
- The RPV was subjected to two hydrostatic pressure tests at 1.25 times design pressure in the works and pre-service.
- The RPV has been subjected to two ten-yearly in-service examinations.

Overall licensee conclusions

5.168. Following a Design Authority assessment of the ageing management processes described above the following conclusions have been determined

1. Sizewell B has a comprehensive programme of surveillance programmes that constitutes the AMP. The arrangements align with those of IGALL and in some cases exceed the requirements.
2. Through the surveillance programmes there is adequate coverage of monitoring the plant operating conditions, a comprehensive in-service inspection programme, sampling of actual RPV steels and testing of these materials. The experience from the programmes and the coverage of ageing mechanisms is considered to be comprehensive.
3. The inspection requirements exceed industry standards and the processes and techniques are validated by third party experts.
4. Further RPV steel specimens have been inserted in the reactor to gain further knowledge of material properties.
5. Processes and arrangements exist to monitor OPEX from the PWR community.
6. EDF-NG is engaged with the OEM (via the PWROG & Framatome Owners Group) and is an active member of EPRI.
7. The integrity of the RPV has been re-affirmed following concerns raised at Creusot-Forge.

5.169. Adequate governance arrangements exist to support ageing management requirements.

Recommendations by EDF-NG

5.170. There are no recommendations raised by EDF-NG against the Sizewell B RPV AMP.

5.2.2 Hinkley Point C (NNB)

5.171. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

5.172. The evolution of the EPR™ design has taken into account feedback from worldwide operating reactors, and specifically from the French N4 and German Konvoi designs. In reviewing the design, and developing the operational practices and surveillance programmes, feedback from EDF SA and EDF Energy surveillance and research programmes has been and will continue to be used.

5.3 Regulator's assessment and conclusions on ageing management of RPVs

5.3.1 Criteria and standards for assessing ageing management of RPVs

5.173. In common with other disciplines, ONR's assessment of structural integrity is based on consideration of established relevant good practice and sound safety principles, the development of which includes ALARP considerations.

5.174. The criteria and standards relevant to this assessment are principally the Safety Assessment Principles (SAPs) (Ref.3), ONR Technical Assessment Guides (TAGs) and ONR Technical Inspection Guides (TIGs), together with relevant national and international standards and relevant good practice informed from existing practices adopted on UK nuclear licensed sites. In addition to more general criteria and standards described in Section 2 of this report, the criteria ONR applies for assessment of ageing management of RPVs are described here.

5.175. The Licensees' submissions to this chapter identify that the SZB RPV is classified by EDF-NG as an "Incredibility of Failure" component; NNB applies an equivalent classification of "High Integrity Component." Both terms reflect the principal role of the RPV in ensuring nuclear safety, and in each case the safety case claims that gross failure of the RPV can be discounted.

5.176. Where such a claim is made, ONR regards these as "highest reliability components" and has particular regulatory expectations. Whilst structural integrity is generally founded on the application of appropriate codes and standards, for highest reliability components ONR expects the safety case to be especially robust. For some aspects, such as the through-life demonstration of defect tolerance, measures that exceed standard design code requirements are necessary in order to satisfy that regulatory expectation.

5.177. ONR's SAPs on integrity of metal components and structures (Group EMC in Ref 3) are relevant to reactor pressure vessels. An ONR TAG on Integrity of Metal Components and Structures (TAG-16, Ref. 45) also addresses this subject. The SAPs and TAG-16 provide guidance on the assessment of ageing management. However their emphasis is on achievement of the structural integrity from start-of-life, which ONR regards as a necessary foundation for ageing management. TAG-16 identifies IAEA standards and guidance applicable to ONR's assessment of structural integrity. Two IAEA publications regarded by ONR as particularly relevant to ageing management of the RPV are IAEA-TECDOC-1503 (Ref. 46) and IAEA-TECDOC-1556 (Ref. 47).

5.3.2 ONR's assessment of EDF-NG's ageing management of the RPV

5.178. Aspects within the scope of EDF-NG's AMP include fatigue monitoring, ISI, water chemistry control, surveillance for boric acid corrosion and a RPV materials surveillance programme. ONR regards the scope of the AMP to be generally consistent with relevant good practice, for example as established under the IGALL programme, therefore it is considered to be broadly acceptable.

5.179. The Sizewell B RPV was designed in accordance with Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (the ASME Code) for a lifetime of 40 years. ISI is conducted in accordance with Section XI of the ASME Code. ONR considers that the ASME Code is internationally well-established and so provides an acceptable basis to demonstrate the high level of integrity required for an RPV.

5.180. Whilst Section III of the ASME Code includes aspects of ageing assessment, the licensee reports additional fracture analysis. This applies the R6 defect assessment procedure (Ref. 48), which accounts for both ductile and brittle failure modes, and includes a conservative estimate of fatigue crack growth through life. ONR considers to this be an acceptable method to establish defect tolerance for the highest reliability components.

5.181. Demonstration of defect tolerance combines the results of fracture assessment with a demonstration of capability for inspections conducted at the assessed locations. The licensee has subjected these inspections to independent validation which ONR regards as providing an appropriate standard for the highest reliability components. A reserve margin, termed Validation Factor by the licensee, is established between a critical defect size and that which can be reliably detected and characterized by inspection. ONR considers this to be an acceptable approach to demonstrate defect tolerance.

5.182. An aspect not discussed earlier in this chapter is that early fracture toughness testing of samples removed from the SZB reactor had indicated that values used in the safety justification may no longer be conservative. In light of this, EDF-NG developed a revised approach that established high capability of detection to maintain a validation factor of 2. This concept was accepted by ONR.

5.183. ONR's assessment is that the methods and outcome of the licensee's ageing assessment of the RPV are broadly acceptable. Application of an established nuclear design code is supplemented by rigorous elastic-plastic fracture analysis. In conjunction with demonstrably reliable inspections, this is considered to provide suitably robust evidence of through-life defect tolerance and is regarded to be a strength in the RPV-specific AMP.

5.184. Whilst it has been identified that monitoring, testing, sampling and inspection are generally conducted in accordance with technical requirements given in Section XI of the ASME Code, there are supplementary provisions for ISI and RPV materials surveillance:

- Volumetric ISI is subject to independent validation including blind trials. ONR regards this validation of ISI as an important strength of the AMP.
- Surveillance for neutron damage is based on pre-irradiation testing of Charpy V-notch, tensile and compact tension specimens and post-irradiation testing of Charpy V-notch, tensile and compact tension fracture mechanics test specimens. A number of specimens have a pre-strain applied to model strain ageing. There is also surveillance for thermal ageing, for which treatment of samples commenced before the RPV entered service.

5.185. ONR considers that EDF-NG's provision for monitoring, testing, sampling and inspection of the RPV adequately reflects the high safety-significance of that component. Notable strengths include the independent validation of volumetric ISI, and the maintenance of a link between ISI objectives and the demonstration of defect tolerance. Data obtained from validated inspection before service provides a benchmark against which the results of ISI are compared. This allows for periodic confirmation, based on highly reliable inspections, that the RPV remains free of structurally significant defects.

5.186. The licensee has identified that surveillance programmes target a range of degradation mechanisms, including both thermal and irradiation embrittlement, boric acid corrosion and fatigue. The boric acid surveillance programme was implemented by the licensee as a remedial measure following detection in October 2000 of leakage at the inner ring of the double O-ring seal at the RPV head. ONR considers that these surveillance programmes adequately reflect experience in the nuclear industry, apply relevant good practice and are satisfactory.

5.187. In November 2017, Sizewell B began its periodic shutdown and during its routine boric acid corrosion monitoring, detected leakage from one of the Steam Generators. ONR is seeking assurance from EDF-NG prior to start up, that all other components (including those that form part of the RPV boundary) are not at risk.

5.188. The licensee has reported ISI of Bottom Mounted Instrument penetrations of the RPV. These regions feature nickel-based alloys, for which there is experience in other similar nuclear power plants of stress corrosion cracking (SCC). Whilst no significant defect indications have been detected by the licensee, ONR supports EDF-NG's decision to continue a regime of speculative ISI where relevant experience indicates susceptibility to degradation. Also with regard to SCC, ONR considers the replacement of the RPV Head in 2006, with a design more resistant to degradation, to be a significant and beneficial preventive measure.

5.189. The licensee has developed a forward-looking AMP in preparation for its application for plant life extension. This includes the introduction of new capsules for surveillance of irradiation embrittlement. Whilst preparations by the licensee continue, no proposal for life extension has so far been submitted for assessment by ONR.

5.190. ONR also regulates EDF-NG's ageing management of the RPV through targeted site interventions, meetings and Periodic Safety Reviews. During these activities, if ONR judges that there are shortfalls, we raise formal issues to ensure that EDF-NG re-establishes a level of compliance that ONR judges to be acceptable.

5.191. The last Periodic Safety Review (performed every ten years by the licensee and permissioned by ONR) was assessed by ONR in 2014 (Ref. 29). ONR's assessment raised several findings. One of these related to the Doel 3 findings of hydrogen flaking flaws found within the RPV shell forgings. EDF-NG has since inspected its RPV shell and found no flaws of significance, or similar to Doel 3. This is discussed further below. Fatigue management was also identified as an area which required further involvement from ONR. This is still a focus area for EDF-NG and ONR.

5.192. During routine periodic shutdowns, EDF-NG performs a series of inspections. ONR is made aware of the inspections being performed through regular meetings with EDF-NG (approximately every 6 months). ONR also performs a site intervention during the statutory shutdown, to gain evidence and satisfaction that these inspections have been performed as planned. At the last statutory shutdown (Spring 2016), EDF-NG inspected many of the RPV internal surfaces, including the RPV shell. The RPV shell inspection was not part of the original inspection proposals for this shutdown; instead EDF-NG planned and deployed this additional inspection in response to ONR and WENRA recommendations following the findings at Doel 3. EDF-NG satisfied itself that it did not have any similar flaws to Doel 3 within the RPV core shell.

5.193. ONR has provided significant challenge to EDF-NG, asking for further evidence that Sizewell B is not affected by the issues from Creusot Forge and Doel 3. EDF-NG has provided (and continues to provide in the case of the Creusot Forge anomalies) such evidence. ONR is satisfied that EDF-NG has replied reasonably to all ONR concerns raised on these matters thus far. However, regulatory engagement on this matter continues and has recently been extended to include international operating experience concerning Kobe Steel forgings.

5.194. Sizewell B has also considered operating experience in terms of the design of the plant. The plant was built later than a number of sister plants in the US and the design has been improved, in particular avoiding the beltline butt weld in the highly irradiated zone of the RPV core shell. There is also additional control, beyond that specified in the applied nuclear design code, specifically on the content of copper and nickel in the RPV welds (applying wider PWR experience and UK practice). Reducing the content of copper and nickel has been found to reduce the potential of deterioration of material properties due to irradiation. Experimental studies at the time of design showed that nickel in combination with copper content can contribute significantly to irradiation embrittlement. ONR is satisfied that EDF-NG took reasonable measures at this time to ensure the potential for irradiation embrittlement was minimised and that it continues to measure these properties suitably through its surveillance programme.

5.3.3 ONR's assessment of NNB's ageing management of the RPV

5.195. Ageing management of the UK EPR™ RPV has been assessed in detail by ONR through its Generic Design Assessment (GDA) process, and is reported in Ref. 49. Examples appear throughout this section.

5.196. The AMP for the UK EPR™ RPV is presently incomplete and its development by the licensee continues. The scope of assessment within GDA was therefore limited. Key aspects of design for ageing management were assessed. These included the location of RPV welds in areas subject to irradiation embrittlement and the detailed specification of RPV materials composition.

5.197. Other aspects were excluded from the scope of ONR's GDA. Matters such as the detailed specification of the RPV materials surveillance and ISI programmes were not considered in GDA and are instead being assessed by ONR during the nuclear site licensing phase, which is underway.

5.198. Consequently it is impracticable to present a complete assessment of a fully developed AMP for the RPV of the UK EPR™ here. What follows instead is a review of its status and maturity, in which key regulatory judgements and observations to date are identified.

5.199. ONR's SAPs state that the positioning of welds should have regard to high-stress locations and adverse environments. Therefore designs should consider avoiding welds in high neutron radiation locations. ONR assessed the acceptability of this design feature at Step 3 of the GDA (Ref. 50) and concluded that it was acceptable based on the following:

- there is a heavy reflector outside the core but within the core barrel which substantially reduces the neutron dose on the RPV wall.

- the end of life neutron dose is similar to that from the highest dose to a weld in a similar design that avoids a weld at this location
- ONR judged the proposed chemical compositions, defined in the RCC-M Code (Ref. 51) for the main forgings, to generally be acceptable.

5.200. The licensee has designed the RPV according to the requirements of the RCC-M Code. ONR found these requirements to be generally equivalent to those of Section III of the ASME Code, with which ONR is familiar and accepts; therefore they were judged to be generally acceptable.

5.201. The proposed fracture mechanics methodology differs from that normally used in the UK nuclear industry and the inspection techniques have some novel features. These aspects were assessed in some depth by ONR (Ref 49). Over the course of GDA the licensee refined its fracture mechanics methodology, based on the French RSE-M code, to develop a technique which ONR has accepted is broadly equivalent to the UK R6 defect assessment procedure which is familiar in the UK. ONR also found there was adequate evidence of capability of manufacturing inspections and of accessibility for in-service inspection.

5.202. The licensee did not undertake lifetime fatigue crack growth (LFCG) predictions as part of its fracture mechanics assessments submitted for GDA. This was because it considered that LFCG is not significant for areas sensitive to fast fracture. Whilst the omission was judged to be acceptable for the purposes of GDA, ONR's expectation is that LFCG should be accounted for in the demonstration of defect tolerance for highest reliability components. Those calculations will be undertaken by the licensee as part of the project-specific detailed design studies when the final site specific loadings have been determined. ONR accepts that this is a suitable approach.

5.203. Ref 49 identifies that ONR's initial assessment was that the original inspection proposals appeared unlikely to be sufficiently targeted to defects of the most likely orientation to be capable of being qualified within the UK. ONR's concerns were discussed during the GDA process and have resulted in proposals for the ferritic welds in the RPV which ONR now considers to be generally satisfactory.

5.204. ONR's assessment is that the methods and outcome of the licensee's ageing assessment of the RPV are broadly acceptable. An avoidance of fracture demonstration has been developed which integrates fracture mechanics analyses, material toughness testing and qualification of manufacturing inspections. ONR considers this to be an acceptable approach to demonstrate, by appropriate margins, the absence of significant defects.

5.205. ONR is of the understanding that fracture toughness testing will be performed on materials subject to a simulated post weld heat treatment to provide a conservative estimation of the fracture toughness.

5.206. The licensee has identified that monitoring, testing, sampling and inspection for the RPV will include the following:

- an irradiation surveillance programme,
- fatigue management through a transient monitoring programme,
- ISI,
- monitoring and control of primary water chemistry,
- periodic review of operating experience and
- a boric acid surveillance programme

5.207. The details of these aspects are still under development by the licensee and so have not been assessed fully by ONR as yet. Whilst ONR is satisfied that the licensee has adequately identified the broad principles of a AMP for the RPV, our detailed regulatory assessment will take place during the current site licensing phase.

5.208. The licensee has identified diverse provisions to minimise the potential for SCC. These include the use of resistant materials, control of water chemistry and the elimination of BMI penetrations. ONR considers these aspects to be a strength of the licensee's AMP established within GDA and discussed within the NNB's submission.

5.209. The licensee presents a fluence-based approach to estimate irradiation damage as the first choice and a displacement per atom (dpa) based approach as secondary, however, this is contrary to the international consensus which is to use dpa as the correlating parameter.

5.210. In GDA, ONR reviewed access for ISI and assessed some of the principles relating to the irradiation surveillance programme. Because ISI will not be carried out until the plant enters service, ONR considers it acceptable that confirmation of its fitness for purpose occurs during the site licensing phase. Ref. 49 notes ONR's expectation that, for the highest reliability components, such confirmation would be through inspection qualification.

5.211. The licensee proposes an RPV materials surveillance scheme using representative samples of parent materials and welds inserted in capsules into a high dose region of the RPV.

5.212. In Ref. 49 ONR accepted that irradiation hardening is likely to dominate the observed rate of embrittlement. However, ONR considered that thermal ageing may be significant for forgings operating at higher temperatures, and Assessment Finding AF-UKEPR-SI-29 was raised. This requires that the licensee shall have access to an adequate database so that thermal ageing effects can be reliably predicted. If necessary, a thermal ageing surveillance programme should be established for materials operating at temperatures experienced by the RPV outlet nozzles. The lack of surveillance for thermal embrittlement, or justification that none is required, is regarded as a potential weakness of NNB's present AMP. However ONR acknowledges the developing status of the licensee's AMP and the matter will be subject to further regulatory assessment during the site licensing phase.

5.3.4 Overall conclusions on ageing management of RPV

5.213. Whilst the AMP of EDF-NG is well developed, that of NNB is less mature. ONR accepts that this difference reflects the different lifetime stages for the two stations being operated.

5.214. For EDF-NG, whilst the licensee does not have a specific ageing management process, it has all the elements necessary within its management system. ONR notes that this is consistent with its regulatory expectations. ONR's assessment of the licensee's contribution to this chapter of the national assessment report demonstrates that the licensee has an adequate AMP for the RPV.

5.215. There is one area of focus recommended by ONR, which EDF-NG is already working on and holds regular discussions on the matter with ONR. This is related to the fatigue management of the RPV. In particular, EDF-NG is concentrating on a detailed understanding of the transient cycles, fatigue usage and keeping up to date with changes to fatigue usage curves, through its continued surveillance monitoring of historical data and predictions to confirm conservatisms in the initial studies.

5.216. NNB is in a position where it has not developed a full AMP, but has the necessary elements of it given that it is in the early stages of licensing and construction. Assurance for ONR comes from GDA. The generic design assessment undertaken by ONR gives it confidence that the design contains suitable provision for ageing management of the RPV

5.217. Although it is still too early for NNB to start considering exactly how some aspects of its ageing management plan will be developed, it is worth highlighting some key focus areas for future regulation of this topic:

- Fatigue in fracture analysis
- Undefined ISI programme
- Undefined surveillance programme
- Thermal ageing

5.218. Acknowledging that Hinkley Point C is a plant in the early stages of construction, ONR considers that the status of the above aspects adequately reflects the maturity of NNB as a licensee. Therefore ONR does not regard these matters to be areas for improvement. Each has been subject to close regulatory scrutiny during GDA, a process which is continuing as part of the site licensing phase.

5.219. No specific areas for improvement have been identified for either licensee.

5.220. The overall conclusion from the assessment of RPVs is that both licensees have adequate AMPs, given the specific stages of the lifecycles of their stations.

6 Calandria/pressure tubes (CANDU)

Not applicable as UK does not operate any CANDU reactors

7 Concrete containment structures

7.1 Description of ageing management programmes for concrete containment structures

7.1.1 Scope of ageing management for concrete containment structures

7.1.1.1 Operating reactors (EDF-NG)

7.1. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

Introduction and description of structures

7.2. The continuous safety of the concrete containment structures is ensured by compliance against the duties and arrangements set out in the Nuclear Site Licence Conditions.

7.3. EDF-NG operates a single PWR power station in the UK at Sizewell B.

7.4. The PWR concrete containment structures (comprising both the primary containment, known as the pre-stressed concrete containment vessel, PCCV, and the secondary containment structure, see Figure 7.1) are subject to routine monitoring and surveillance as a key part of the overall ageing management strategy. The containment structures are required to remain safe to operate for their lifetime requirements and timely surveillance helps to identify any material condition shortfalls that could preclude this objective from being achieved.

7.5. The AMP ensures that the safety functionality and duties of the containment structures are not affected by ageing mechanisms during the lifecycle of the station.

7.6. All of the elements that make up an AMP for the Sizewell B concrete containment structures are either embedded in the station safety case or are part of company procedures and processes. The general ageing management arrangements are discussed in Chapter 2, and will not be repeated here. This chapter will focus on the specific arrangements for the concrete containment structures.

7.7. The Maintain Design Integrity (MDI) process controls the modification, maintenance and documentation of the safety case. It ensures that the safety case claims on the concrete containment structures and associated documentation remain consistent with each other and with the design intent. This includes the specification of the surveillance programme, which covers testing and inspection of the structures in order to demonstrate that their condition remains aligned with the safety case. The focus of the commentary in Section 7 is therefore on the surveillance arrangements for the ageing management of the PCCV.

7.8. The supporting analysis to the safety case assesses the structural ageing and degradation in setting appropriate operating limits and the inspection and testing regimes. For significant lifetime issues this may also include identification of appropriate research and development to underwrite ageing and degradation and other modelling assumptions.

7.9. Within EDF-NG, the design authority role is carried out by the Design Authority organisation (under the Head of Design Authority). Several other parts of the licensee organisation may be required to support the Design Authority in the delivery of its role e.g. Plant Discipline SQEP(s) where specific technical knowledge and authoritative advice is required (and resides out with the Design Authority). This arrangement is embedded within a Company Specification.

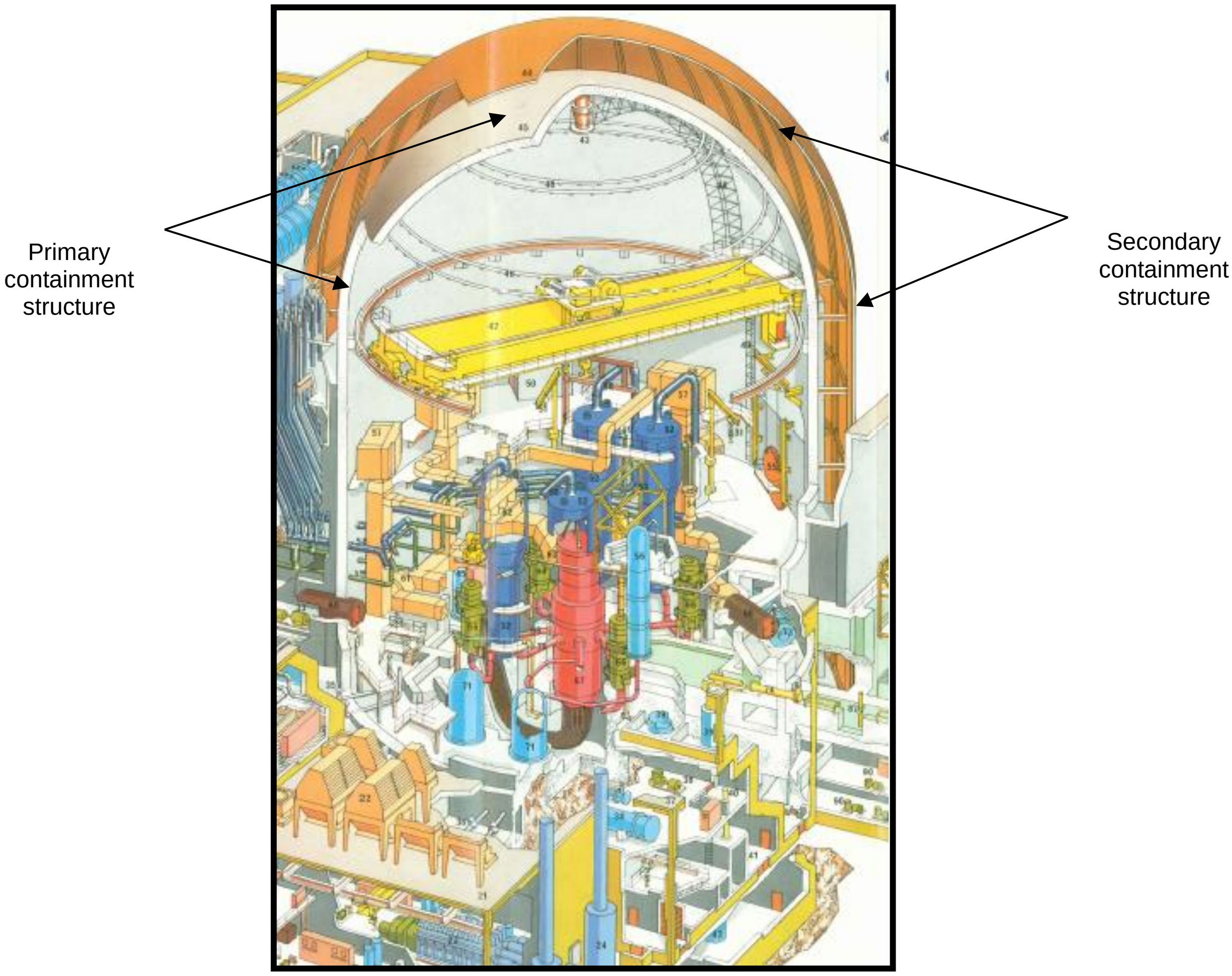


Figure 7.1 Cut-away cross section of the PWR containment structures.

7.10. In the case of the PCCV structure, the PCCV Design SQEP role is enacted by civil engineering SQEPs to provide technical and authoritative advice for safety issues affecting the PCCV safety case, integrity, operation, review etc. Proposed changes to the design envelope, operating function or monitoring of the PCCV must be undertaken in accordance with the safety case modifications process.

7.11. Authoritative technical advice on the PCCV is provided by SQEP personnel who can fulfil the PCCV Design SQEP role and subject to independent review and agreement by the nominated Appointed Examiner. Any structural defects, non-compliances and/or ageing issues that may be identified by the Appointed Examiner are subject to review and assessment by the PCCV SQEP in accordance with the company governance arrangements; this ensures that ageing issues that could affect the safety case envelope are adequately addressed.

7.12. PSRs are carried out every ten years in accordance with the licence condition for periodic review (paragraph 2.48). For the PWR containment structures there have been two such full reviews to date with the current PSR report(s) providing confidence that the PCCV and secondary containment continue to be safe as judged against the safety standards and practices in force at the time. The main objectives of the PSR reviews are to:

- Validate that the physical conditions and capabilities of the primary and secondary containment structures remain in accordance with the Safety Case requirements. This includes the structural integrity under both normal operational conditions, and against the identified internal and external hazards.
- Confirm the adequacy of the primary and secondary containment structures for operation at least for the period until completion of the next PSR.
- Review relevant developments in research and development activities, and national and international experience.
- Validate that any ageing and degradation mechanisms or other factors that could affect future safe operation are reflected within the safety case, or that measures are in place to ensure they are duly considered.

These objectives align with the requirements of AMP.

7.13. Looking ahead to PSR3, the review strategy will focus more upon challenging the generic arrangements on how the SSCs are managed. This will primarily consider the effectiveness of technical governance, plant health committees, peer group improvement plans, equipment reliability reviews (including SSRs), system action plans and the maintenance schedule adequacy. Hence the integrated processes that are embedded across the business will be reviewed in detail to assess their continued contribution to the overall reliability improvement for the SSCs. This approach still aligns with the broader AMP remit whereby many of the key initiatives (e.g. TLMS) and decision influencing (e.g. risk management) become embedded into normal business.

7.14. Due to the passive nature of the concrete containment structures (similar to many large civil structures), the AMP surveillance and condition trending activities do not strictly align with the company Equipment Reliability (ER) processes. The ER process is focused predominantly upon plant and system components where active changes in plant configuration & performance can be readily measured and recourse to condition monitoring methods can be used to generate informative metrics. The long term condition and trending of the concrete containment structures is included in the existing SHIP processes but the output offers only a limited overview of the actual structural performance and understanding required to provide relevant information to inform forward AMP strategies. Instead, the long term performance of the containment structures is more effectively informed by the detailed surveillance programmes and associated reporting. This alternative approach more than counters any shortfalls in trying to fully align a plant orientated ER process to large civil structures.

7.15. EDF-NG implements Technical Governance (TG) arrangements for the containment structures to provide assurance that these critical structures are appropriately managed. A top level company strategic policy for the PWR Containment is supported by a mandatory company technical standard for management of the concrete pressure vessels. This specifies the requirements for safe management of the PCCV and sets down the appropriate governance arrangements. The governance of the containment design codes is addressed via a technical guidance document which sets down guidance for the use of civil engineering design codes, including ASME III as used in modified form for the original PCCV design and ASME XI which specifies the in-service inspection requirements.

7.16. Risks associated with the containment structures are addressed in the same manner as all other plant risks via the company risk log and risk management process. This ensures that civil engineering risks associated with the concrete containment structures are identified, understood, valued and appropriate mitigation strategies are developed. The risk profile for the concrete containment structures is continuously monitored and reported to senior and executive management. The risk structure for the PWR Containment Structures is primarily represented by the following top level risks (which act as headings to organise the significance and relevance of potential risks):

- Failure of liner and penetrations.
- Failure of tendons/ stressing system.
- Failure of concrete.
- Failure of foundations.

7.17. The risk values both in nuclear safety significance and commercial risk are reviewed quarterly throughout the year and the current risk position and mitigation progress is reviewed and reported to senior management. The risk management process aligns with the ageing management principles to ensure visibility of life limiting issues and to support investment.

7.18. The wider remit of governance across the fleet for critical plant items includes the establishment and ownership of a Through Life Management Strategy (TLMS). A TLMS has been developed for the PCCV and the strategy focuses on the components most vulnerable to ageing related degradation. The TLMS systematically identifies the activities needed to achieve the lifetime requirements of the structure and associated safety components. The top level aims of the TLMS are to:

- Collect together a number and range of PCCV related initiatives and strategies under a ‘single banner’.
- Develop a standardised approach to safely achieve lifetime requirements.
- Provide senior management and the investment decision makers with visibility and an understanding of the investment landscape.
- Develop a methodology to prioritise investment decisions that support ageing management.

7.19. The PCCV strategy is divided in three sections:

- Reviewing the adequacy of the current maintenance, inspection and testing regimes.
- Development of performance improvements.
- Support resolution / mitigation of plant issues.

7.20. The maintenance, inspection and testing is considered to be discharged as part of ‘normal business’ and provides the foundation for the overall ageing management process. The remaining two sections are of equal importance to the overall lifecycle requirements of the PCCV.

7.21. Key areas for specific monitoring arising from the PWR TLMS to date include:

- Monitoring for groundwater ingress that could in the future threaten the lower liner integrity.

- Pre-stressing tendon loads approaching lower design limits and associated safety case issues.
- Monitoring the risk of high concrete temperatures around steam/feed penetrations.
- Sump drain liner thickness degradation (where exposed to boric acid corrosion).

7.22. The primary containment (known as the Pre-stressed Concrete Containment Vessel, PCCV) takes the general form of a pre-stressed and reinforced concrete cylinder and hemispherical dome founded on a reinforced concrete base. The cylinder shell is surmounted by a hemispherical dome (Figure 7.2).

7.23. The reinforced concrete base is essentially unpenetrated but has a central depression to accommodate reactor instrumentation lines. An annular pre-stressing gallery is located around the periphery of the underside of the base. Major penetrations are located in the cylindrical wall of the containment where the major advantage of pre-stressed concrete ensures that this part of the structure and the dome are essentially free from through thickness cracks under accident conditions.

7.24. The containment is sized so that the net free volume limits the internal pressure rise to a design pressure of 0.345MPa (maximum fault pressure + 10% margin). The foundation is a reinforced concrete slab. The base slab is locally stepped to form a lined central cavity that accommodates the Reactor Pressure Vessel (RPV) and associated instrumentation cables.

7.25. The principal safety functions of the Primary Containment (PCCV) are:

- To house and support enclosed plant in a suitable environment in combination with the heating, ventilation and air-conditioning system.
- To provide support to the secondary containment.
- To protect plant and equipment from external events.
- To provide and maintain radiological shielding.
- In conjunction with the reactor pressure vessel and other systems, to prevent the uncontrolled release to the environment of fission products.
- To withstand the effects of normal and exceptional load combinations to ensure that adjacent structures are not precluded from fulfilling their safety function & duty.

7.26. The complete internal surface of the primary containment is covered with a mild steel liner to ensure leak tightness. The base slab liner is covered by an additional reinforced concrete floor slab, providing a finished ground floor level. The boundary of the primary containment structure forms the extent of the Reactor Building.

7.27. Electrical and mechanical services pass through the liner via specially-designed containment penetrations. Access for personnel and materiel is achieved through two Personnel Access Airlocks and an Equipment Access Hatch.

7.28. Fuel movements between the Reactor Building and the Fuel Building take place through a special penetration system, the Fuel Transfer Tube. This liner, the penetration assemblies and the containment isolation provisions constitute the primary containment boundary.

7.29. A secondary containment structure provides the environmental protection to the primary containment PCCV where the primary containment wall or dome would otherwise constitute the outer building boundary. There is an interspace between the two concrete structures that is maintained below atmospheric pressure. By this means any leakage from the PCCV would be contained.

7.30. The secondary containment structure is constructed of lightweight reinforced concrete and the upper dome portion is supported on cantilevers which project from the primary containment near the top of the cylinder wall. The secondary containment structure continues to ground level on one partial side of the primary containment where there are no immediately adjacent buildings (with the exception of the main equipment access hatch).

7.31. On the other side, the adjoining rooms of the Auxiliary Building and Fuel Building provide the environmental protection to the primary containment wall (Figures 7.2, 7.3 & 7.4). In these areas, any leakage via the primary containment buttress wall is managed via a ventilation filter system.

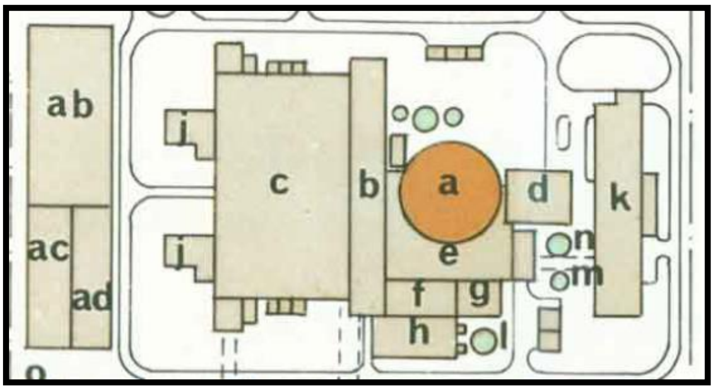


Figure 7.2 Plan of main buildings around the primary containment.

7.32. These adjoining buildings comprise conventional reinforced concrete construction and, by default, the rooms immediately adjacent to the primary containment wall form the secondary containment boundary in the event of any leakage. The primary containment and the adjoining buildings are notionally separate structures with the seals and joints between forming part of the containment.

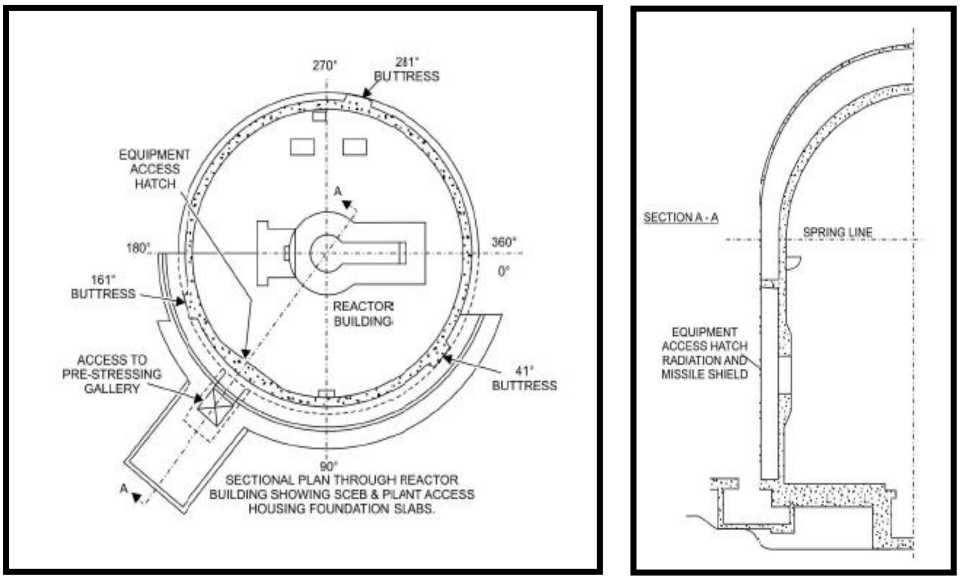


Figure 7.3 Plan and cross section of reactor building.

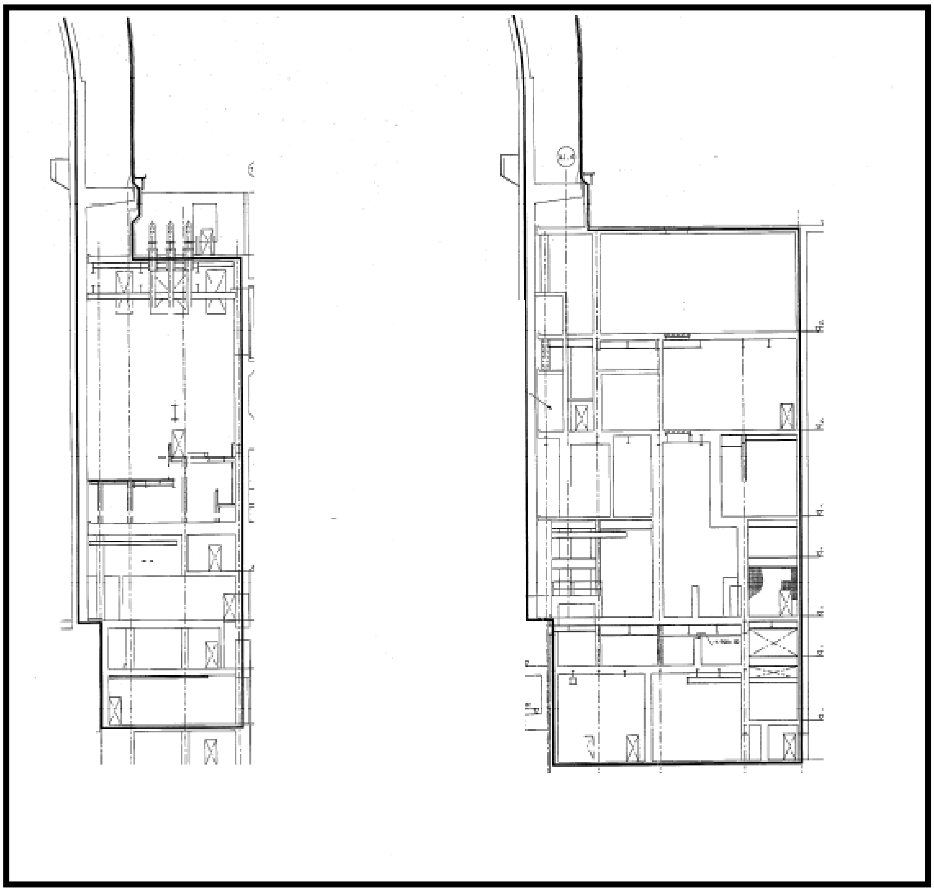


Figure 7.4 Sections of reactor building & auxiliary building showing extent of the containment boundary.

Methods and criteria for selecting components within the scope of the AMP

7.33. The PCCV is required to maintain structural performance over required levels (defined in the safety case) with adequate reliability during the design service life. Consequently, inspection and maintenance are an indispensable part of the AMP to ensure the on-going safety and integrity. The AMP encompasses a multi-activity approach that includes inspection, recording, trending, degradation prediction, evaluation, judgement and remediation.

7.34. The PCCV constitutes a massive civil structure whose performance and duty is largely passive and predicated upon maintaining an adequate margin against failure for all credible loading scenarios including faults and hazards. The structure comprises of significant structural components that combine to provide the overall robustness and strength of the PCCV. Failure of these components including due to ageing has the potential to reduce the safety margin. The scope of the PCCV AMP is therefore informed on this basis to identify the structurally significant components and their degradation mechanisms.

7.35. The ASME inspection rules for concrete containment underwrites this approach and recommends specific surveillance activities that support the on-going evaluation of the structure and any deterioration including observation of deflections, cracking, temperatures and cracking – to be periodically monitored for compliance against the design basis assumptions.

7.36. The scope of the AMP in this section covers the following key aspects:

- Concrete structures.
- Steel reinforcement.
- Pre-stressing systems.
- Liner.
- Boric acid effects (including coatings, steel surfaces).
- Interactions of Liner and Concrete Containment (including Equipment Hatch and penetrations).
- Seals (moisture barrier at barrel to basemat junction).

7.37. These key elements are hereafter used as representative components of the concrete containment structures to form a 'golden thread' of examples.

Processes/Procedures for the identification of ageing mechanisms

7.38. A key element of the AMP is to undertake condition monitoring and inspection of the concrete containment structures at frequencies and using methods that are appropriate to the importance of the structure to nuclear safety.

7.39. With regard to the condition monitoring of the containment structures, a distinction may be drawn between this specialised form of surveillance and other types of structural survey. It is therefore useful to define these as follows:

- Condition monitoring: In-service surveillance of structures on a continuous or quasi-continuous basis using imbedded or externally applied monitoring devices. This includes automated systems linked to data-logging equipment which may have pre-defined alarm levels.
- In-service inspection: In-service surveillance of structures by periodic surveys at specified intervals. This may include consideration of information obtained during condition monitoring surveys over the period between inspections.

7.40. In-service inspection of the PWR containment structures and other nuclear safety-related civil structures is specified in accordance with the station specific 'Rules for In-Service Inspection of the Civil Engineering Works' which defines the scope, frequency, and acceptance standards for these activities.

7.41. The "Rules" include:

- Examination of the PCCV as covered by the PWR Technical Specification Surveillance Programme.
- The PWR Containment Leakage Rate Testing Surveillance Programme.
- Inspection of other nuclear safety-related civil engineering structures.

7.42. The procedure adopted by EDF-NG in compiling the statutory report on the containment structures is that the report is prepared by the Appointed Examiner (APEX) who is a nominated suitably qualified and experienced (SQEP) chartered civil or structural engineer responsible for the implementation of the monitoring program and the assessment and reporting of the results which culminate in a final report.

7.43. As a key requirement of the role, the APEX provides a SQEP surveillance and inspection capability that is independent of the PWR operation and safety case justification.

7.44. The requirement for this level of independence arises from the following:

- To submit an autonomous SQEP report (independent of station and the corporate “Design Authority”) on the containment structures detailing the results and findings from the various surveillance activities specified in the station surveillance programme (akin to an independent technical audit of the containment integrity against pre-determined safety case requirements for continued operation).
- To provide an independent oversight role, in recognition of the structural importance and unique form of the PCCV design and construction in relation to its nuclear safety significance.

7.45. As the APEX is a direct employee of EDF-NG, there has to be a suitable degree of independence from the functions of the company organisation that relate to the operation of the PCCVs under their surveillance jurisdiction i.e. the APEX is precluded from fulfilling the Responsible Designer function as defined within INSAG-19.

7.46. The Appointed Examiner has a high level of civil engineering technical expertise, particularly with regard to the design, construction and ongoing behaviour of large concrete structures such as PWR PCCVs and the AGR fleet PCPVs. Experience of structural monitoring augmented by specific on the job PCCV training plus plant familiarisation is essential covering such topics as :-

- Overview of PWR Design & Operation.
- Design philosophy of the PWR PCCV.
- Familiarisation with the conditions of the Nuclear Site Licence.
- Familiarisation with station technical surveillance programmes.
- Station Safety case.
- Case history and lifetime records.
- Quality Assurance procedures.
- ASME design rules for containment & civil works.
- Safety training (including pre-stressing supervision).
- Plant familiarisation.

7.1.1.2 Hinkley Point C (NNB

Introduction and description of structures

7.47. This section divides the EPR^{TM} UK HPC containment structures into three parts:

- (i) Reactor Building Inner Containment
- (ii) Airplane crash shell and outer containment
- (iii) Common Raft

7.48. Despite performing different functions the civil engineering requirements applicable to these parts and dealing with the structures lifetime are identical. They are then presented now.

7.49. The civil engineering design must ensure that the structures perform their safety function throughout the planned life of the plant. EPR^{TM} UK HPC structures are designed with an operating life time of 60 years. Additional periods of 5 years for construction and 15 years for decommissioning must also be taken into account.

7.50. The detailed design of the reinforced concrete and steel structures against the load cases defined in AFCEN ETC-C 2010 shall be carried out for a construction and station operating life time of 65 years for the Reactor Building Inner Containment.

7.51. To allow for a decommissioning period of 15 years plus contingency, the durability of reinforced concrete is considered at the design stage with a 100 years' lifespan. Eurocode 2 structural class S5 is then retained.

7.52. It is mentioned below that these structures are Safety Class 1. It shall be noted that Safety Class 1 is assigned to structures whose function is to provide protection against external hazards (including earthquake conditions) or internal hazards (e.g. internal explosion) for:

- Safety class 1 or safety class 2 components involved in a first line of protection,
- Safety classified components involved in another level of defence in depth, if their robustness against earthquake is required.

(i) <u>Reactor Building Inner Containment</u>

7.53. The Reactor Building Inner Containment constitutes the third containment barrier and contributes as such to the confinement safety function.

7.54. The Reactor Building Inner Containment provides a physical, resistant and leak tight barrier that ensures, in combination with associated circuits, the confinement of radioactive material that could be released in all normal, exceptional and accidental situations considered in the HPC EPR^{TM} design.

7.55. The Reactor Building Inner Containment directly contributes to the main safety function of radioactive material confinement.

7.56. The following main safety functions and their associated Plant Level Safety Functions must be fulfilled by systems housed inside the Reactor Building Inner Containment:

- Fuel heat removal
- Control of fuel reactivity

7.57. The Reactor Building Inner Containment is a Safety Class 1 (C1) civil structure in view of its requirement to provide the main safety function of confinement of radioactive material.

7.58. The pre-stressed reinforced concrete Reactor Building Inner Containment is comprised, from bottom to top (see Figure 7.5), of a:

- Cylindrical gusset,
- Truncated section,
- Cylindrical section,

- Torispherical dome connected to the top of the inner containment wall (typical thickness, 1 m). The concrete intersection between the cylindrical wall and dome is called the dome ring.

It includes (See Figure 7.1):

- A leak-tight steel liner on the inner face, anchored to the concrete,
- Support brackets for the polar crane girder,
- Three vertical ribs on the outer face, for anchoring the horizontal pre-stressing tendons,
- Bosses and strengtheners around the transfer tube sleeve and equipment hatch.

7.59. The inner containment wall and the dome are pre-stressed concrete structures.

Pre-stressing is provided by an arrangement of fully cement grouted bonded steel tendons. The tendon characteristics are given in the ETC-C.

- Each horizontal tendon makes a complete loop of the containment and is anchored within one of the inner containment wall ribs. Each horizontal tendon is tensioned at both ends, at the same rib,
- The vertical tendons are divided into two families:
  - Short vertical tendons: tensioned at their lower end located inside the pre-stressing gallery which is located underneath the Common Foundation Raft, and passively anchored at their upper end. This upper end is embedded within the Reactor Building Inner Containment wall,
  - Long vertical tendons: tensioned at their lower end located in the pre-stressing gallery and are passively anchored in the dome ring,
- The “gamma” tendons are vertical tendons which extend into the dome and are tensioned at both ends. The lower end is anchored in the pre-stressing gallery and the upper end is anchored at the opposite side of the dome ring.
- The cable quantities are summarised below:
  - Horizontal tendons: 119,
  - Short vertical tendons: 30,
  - Long vertical tendons: 29 (of which 4 are instrumented and un-bonded),
  - Gamma tendons: 104.
- Other design features include:
  - 54T15 very low relaxation tendons consisting of 54 strands composed of 7 wires (nominal section: 150mm^2)
  - Designed to withstand P= 0.65MPa (abs) and designed to withstand LOCA (rupture of the surge line nozzle: pressure = 0.48MPa (abs)) combined with the load due to the Design Basis Earthquake

7.60. The selection of grouted tendons is substantiated by the 3 following aspects:

- Considerable OPEX from 58 EDF Nuclear Power Plants operating in France,

- The long term protection afforded to the tendons by the alkaline grout, which protects the steel strands and end caps from corrosion, and prevents the ingress of water or other aggressive media,
- In the event of tendon rupture the bonds between the strands and the grout and the grout and the concrete wall allow part of the pre-stress to continue to be transmitted to the structure.

7.61. Regarding the resilience to tendon failure, the pre-stressed containment structure has redundancy i.e. it is tolerant to multiple tendon failures.

7.62. A theoretical study has been carried out to determine the number of hypothetical tendon ruptures that could be tolerated without loss of resistance of the containment structure to the design loads. In addition, the Reactor Building Inner Containment monitoring system (EAU) permits detection of hypothetical tendon ruptures (see section 7.1.3.2).

7.63. The steel liner comprises steel plate sections, welded together, covering the entire internal surface of the inner containment walls, dome and Common Foundation Raft. To ensure continuity the liner base is located between the top of the Common Foundation Raft and the underside of the support slab for the Internal Structures.

7.64. A continuous anchorage system is welded to the steel liner plates and is integrated into the concrete. It comprises continuous vertical and horizontal steel anchors. Inside the areas enclosed by the crossing continuous anchors are meshes of smaller stud anchors. The role of the anchoring system is to stiffen the liner and to ensure its strength during construction and operation.

7.65. The continuous anchors transmit concrete deformation to the steel liner. They limit the movement of the liner relative to the concrete due to differences of thickness, temperature conditions or elasto-plastic conditions in the liner. In addition, the continuous steel anchors provide the liner with sufficient rigidity during its assembly and during the construction phase.

7.66. The spacing of the stud anchorages is such that local buckling, which can occur (due to geometrical manufacturing defects) during pre-stressing or when heated, remains within acceptable limits.

(ii) <u>Airplane crash shell and outer containment</u>

7.67. The main safety functions relative to the APC Shell including the Outer Containment is the contribution to the “confinement of radioactive material”. The APC Shell including the Outer Containment, protects the general public, workers and the environment from normal and accident situations by providing a barrier to the release of radioactivity, the APC Shell including the Outer Containment also provides protection to structures, systems and components housed within it.

7.68. The APC Shell including the Outer Containment houses and protects buildings (Reactor Building, Safeguards Building 1 and 4 and Fuel Building) which house and protect themselves multiple systems and components responsible for the main level safety functions of: removal of heat, control of radioactivity of fuel, and confinement of radioactivity. Alongside the aforementioned systems, the APC Shell including the Outer Containment structure itself contributes to provide a barrier to the control of release of radioactivity into the environment.

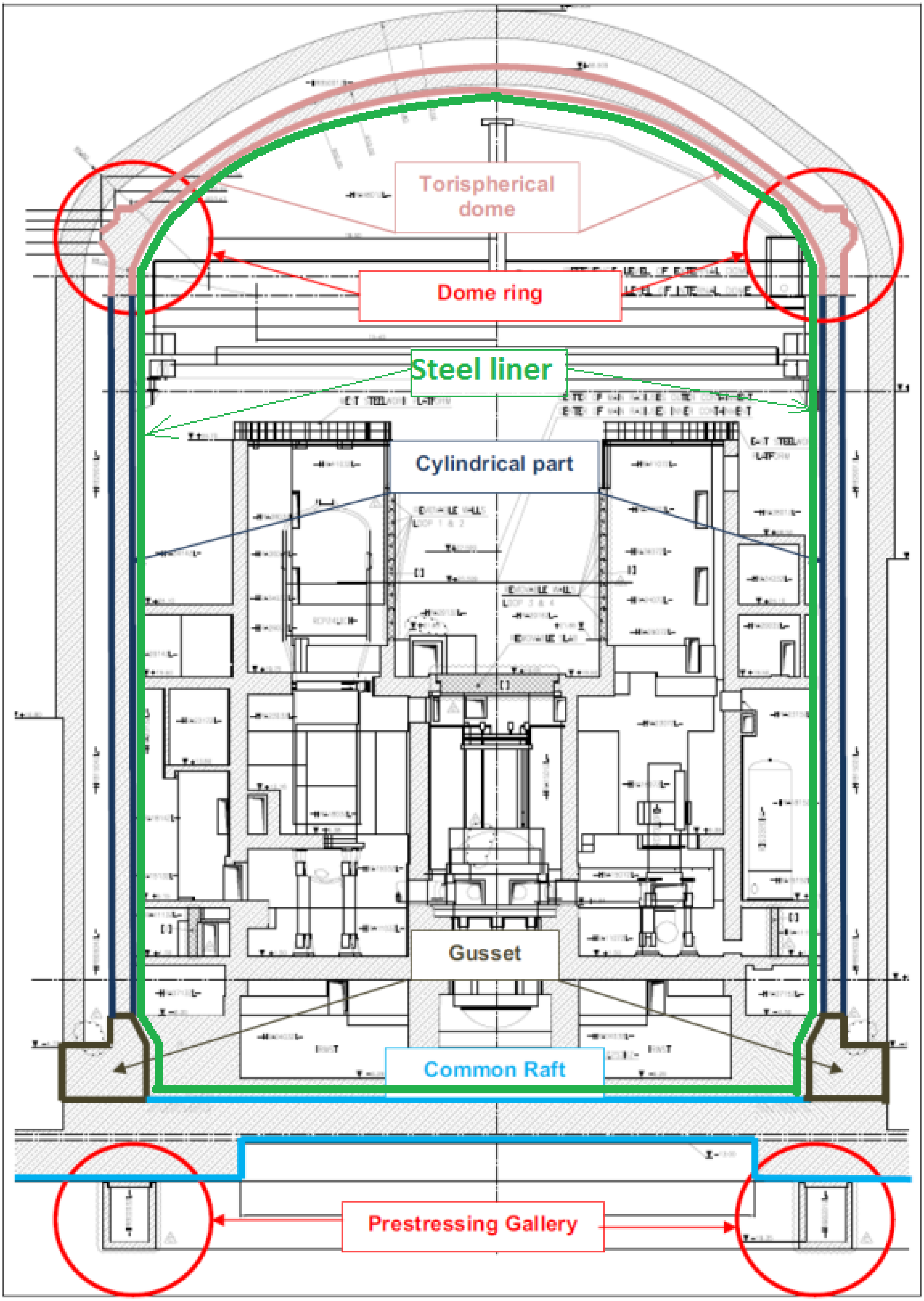


Figure 7.5: Vertical section through Inner and Outer Containments

7.69. The safety classification of the APC Shell including the Outer Containment Is Safety Class 1 as:

- it contributes directly to the main safety function of confinement of radioactive material (identified as barrier role for civil structures)
- it protects and houses components with a containment barrier role,
- it protects and houses components involved in the first line of protection and therefore classified Safety Class 1 or Safety Class 2.

7.70. The global dimensions of the Outer Containment are:

- Internal diameter: 53 m,
- Height: 49.50 m (cylinder without dome)
- Dome: from level +45.15 m OD to +62.31 m OD.

7.71. The containment annulus is located between the inner and the outer containment. The annulus ventilation system:

- Maintains the annulus at a negative pressure to collect any leaks from inside the containment following an accident.
- Discharges these leaks to the vent stack after using High Efficiency Particulate Air and iodine filters.

(iii) Common Raft

7.72. The EPR^{TM} UK HPC Common Raft is a reinforced concrete structure, in the shape of a cross within a square, which supports the Reactor Building, the safeguard buildings (HL), the Fuel Building and the APC shell. Loads are transferred from the slabs to walls at each level all the way down to foundation level.

7.73. The Common Raft fulfils a dual role. On the one hand, it must safeguard the classified safety systems/components it houses against all the fault and hazard conditions to which they might be exposed, in particular external hazards. On the other hand, it must protect the environment against all the fault conditions whose frequency of occurrence cannot be reduced to insignificant levels. It must limit any release of radioactive substances.

7.74. The safety classification of the Common Raft is Safety Class 1 (C1) as it houses equipment and systems that fulfil a Category A function and also houses fuel and components containing radioactive material.

Methods and criteria used for selecting components within the scope of ageing management

7.75. Based on the design specifications (e.g. 100 years required lifetime for reinforced concrete, maximum leak rate for waterstops) and considering the HPC specific conditions (e.g. seawater, soil aggressiveness) the various materials are selected in compliance with adequate standards (e.g. EN206, BS8500).

Processes/Procedures for identification of ageing mechanisms

7.76. The design of the Reactor Building Inner and Outer Containments is still under development and will ensure that such maintenance activities are achievable. The building layout will be such that systems and components are accessible and that it is safe and practical to undertake maintenance. The ageing management programme will be developed as construction progresses.

7.77. It can nevertheless be noted that inspection typically includes:

- Visual inspection of the concrete surfaces, with the recording and monitoring of any defects such as significant cracks in the concrete surfaces. Non-structural panels and equipment are designed to be removable as required to facilitate inspection of the wall surfaces.
- Visual inspection of seismic gaps with adjacent structures to make sure they are clear of debris.
- Visual inspection of fire stopping to service penetrations through the concrete structure.
- Visual inspection to record any signs of water leakage through the concrete structure or the liners of pools, tanks or sumps
- Monitoring of the water leakage detection system to control liner and sumps

The inspection will be undertaken as continuous processes as part of normal business.

7.78. The plan will require that defects are recorded and categorised in terms of their nuclear safety significance. It is expected that the maintenance requirements for the concrete structure is minimal, given that the reinforced concrete is designed to be durable for an 85 year life span. However, maintenance issues may arise from defects recorded during the periodic inspections but, provided that defects are repaired in a timely manner, the structure is expected to maintain its safety functions over the planned 85 year design life.

7.79. In addition, and as part of the periodic safety reviews undertaken every ten years during the Nuclear Power Plant (NPP) lifetime, a complete assessment of ageing and any required major maintenance activities are performed to confirm that the Reactor Building Outer and Inner Containments structural performance continues to meet its design requirements and that the plant is suitable for operation for a further 10 years' period.

7.80. Further maintenance activities on the SSCs within Reactor Building and on Reactor Building Outer and Inner Containments structure itself will be undertaken during outage periods.

7.1.2 Ageing assessment of concrete structures

7.1.2.1 Operating reactors (EDF-NG)

7.81. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

7.82. The ageing assessment strategy has been developed based upon a range of key design documents and safety standards which focus the surveillance activities upon potential issues at greater risk of ageing degradation. In addition, the recourse to OPEX provides insight into other unforeseen risks that could manifest with time.

7.83. The AMP for concrete containment structures for each of the NAR examples is now discussed.

(i) <u>Concrete structures</u>

7.84. By careful mix of design and quality of construction, the concrete containment structures have not exhibited any adverse evidence of internal chemical reaction leading to material degradation or incompatibility with the immediate operating environment.

7.85. The main ageing issues relate to monitoring the structural performance of the containment structures. The ageing effects and mechanisms differ between the buildings and structural components depending upon: (i) the form and materials of construction; (ii) their location e.g. sheltered, exposed, buried foundations, etc.; (iii) type of load actions to be resisted during normal operating and faults (by default some load actions occur and act routinely on the structure whilst other may be exceptional loads based upon fault conditions for which the structure is designed to resist in the unlikely event of occurrence). The AMP is intended to detect early indications of material degradation that could affect the structural performance (and hence safety duty) of the structural component under review.

7.86. The predominant feature of concern would be evidence of concrete cracking and hence much of the AMP surveillance is focused on detecting such behaviour. Categorisation of concrete defects is established as part of the technical surveillance programme that covers the containment in-service inspection.

7.87. As per the guidance of Section IWL of ASME XI, the condition of the concrete surfaces is acceptable if the Appointed Examiner determines that there is no evidence of damage or degradation sufficient to warrant further evaluation or performance of repair/replacement activities.

(ii) <u>Steel reinforcement</u>

7.88. The embedded steel reinforcement is generally judged to remain in good condition (and hence capable of delivering its structural duty) as long as the surrounding concrete matrix offers a non-corrosive environment (notionally alkaline) and provides a source of passivation protection.

7.89. In the absence of wetting or exposure to chlorination / carbonation, the overall risk to the reinforcement is judged low. The primary containment is entirely sheltered from the environment by the secondary containment and hence has very low exposure to carbonation effects form the atmosphere and chloride ingress due to the marine environment. The secondary containment is also clad with protective weather tiles and has limited exposure to external conditions.

7.90. The acceptance criteria are closely linked to assessing the visible condition of the concrete surface whereby any effects from steel reinforcement degradation would manifest. Where the concrete surface is steel lined (e.g. inner containment faces) the effect of localised reinforcement corrosion would be less obvious as the steel liner plate would obscure the initial distress. In such circumstances, the steel liner would exhibit evidence of bulging, blistering or corrosion albeit at a slower rate.

7.91. OPEX from the United States confirms that visual distress to the steel liner occurs when construction detritus (e.g. small sections of timber), is erroneously left within the concrete pour and thereafter degrades.

7.92. In the event of wider steel reinforcement corrosion occurring, it is judged unlikely that the outer exposed concrete face would not show such evidence and, as above, the steel liner would exhibit areas of distress.

(iii) <u>Pre-stressing systems (Primary Containment only)</u>

7.93. The major ageing issues for the pre-stressing system are:

- reducing tendon loads;
- tendon condition; and,
- anchorage condition.

7.94. These comprise the critical components of the pre-stressing system and maintaining them in an acceptable condition is a safety significant activity for the continued safe operation of the containment and identification of longer term ageing mechanisms.

7.95. Tendon load checks on the un-bonded pre-stressing help to verify the residual pre-stressing load in the PCCV to provide an early detection of any adverse loss of load and to assist with long term forecasting.

7.96. The acceptance criterion is based a comparison of the residual (as measured) tendon loads against the minimum pre-stress design level as specified in the station safety case as follows:

(a) The average of all measured tendons ≥ minimum required pre-stress specified at the anchorage.

(b) The measured force in an individual tendon ≥ the predicted force.

(c) The pre-stressing forces and measurements from previous examination should indicate a pre-stress loss such that predicted tendon forces will meet the minimum design pre-stress at the next scheduled examination.

(d) Variation of measured tendon elongation from the last measurement, adjusted for effective wires or strands, should be by less than 10%.

In the event that criterion (b) is not satisfied, Sub-section IWL of ASME XI specifies additional measures that need to be demonstrated.

7.97. Strands are removed from a number of tendons for visual examination for signs of deterioration in their protective coating and underlying corrosion or mechanical damage. These are tested to determine their mechanical properties and cumulative records of the as-found condition are retained in a statistical format for comparison against permissible levels.

7.98. The condition of the strands is acceptable if:

(a) Samples are free of physical damage.

(b) The tested values of ultimate tensile strength and elongation are not less than minimum specified values.

7.99. Tendon anchorages are inspected for evidence of slippage, damage, corrosion or other signs of general deterioration. The condition of the anchorages (covering both the tendon anchorage components, bearing plates and adjacent structural concrete) is acceptable if the Appointed Examiner determines that there is no evidence of damage or degradation, leakage of corrosion protection medium, or end-cap degradation sufficient to warrant further evaluation.

(iv) <u>Liner</u>

7.100. The main liner is the principal component of the containment boundary and the Technical Surveillance Programme requires that it is inspected to ensure it retains its structural integrity, and hence leak tightness.

7.101. The mild steel liner is coated with protective paint which was selected to provide corrosion protection during normal operations and to survive under fault conditions. The coating protects the liner from oxidation and corrosion as long as it remains intact. Damage to the coating could expose areas of the liner leading to oxidation of the exposed metal. A key part of the AMP considers the ageing mechanism for the lower parts of the liner due to attack by boric acid solution, if exposure to the primary coolant was to occur and the liner coating was damaged.

7.102. Furthermore the AMP focuses on the movement joints and moisture barrier in the region where the liner plate joins the floor. The joints were well designed and installed in accordance with a rigorous QA/QC programme during the original construction. No indications of degradation due to concrete placement deficiencies in this vicinity of the liner have been noted.

7.103. At construction, the welds between the plates making up the floor liner were vacuum box tested and then covered by leak-chase channels with vertical upstands fitted at each end of the run of leak chase channel, to allow pressure testing after the internal floor concrete was placed. The AMP recognises the potential for corrosion to develop at the leak chase channels and these are subject to detailed inspection.

7.104. The surveillance requires visual examination to be undertaken, either directly or remotely, by line of sight from permanent vantage points only or unless temporary access is stipulated by the inspection plan. All visual examinations are undertaken in accordance with ASME XI Subsection IWE 2300 to “General Visual Examination” standards with adequate illumination sufficient to detect evidence of degradation.

7.105. The surveillance covers the condition of the liner and protective coating to check for evidence of flaking, blistering, peeling, discoloration and other signs of distress.

7.106. The acceptance criteria are based upon AMSE XI Subsection 3500 whereby the Appointed Examiner determines whether the condition is acceptable and there is no evidence of damage or degradation sufficient to warrant further evaluation or performance of a repair/replacement activity. Suspect conditions are evaluated to the extent necessary to ensure the component function is not impaired.

7.107. The liner plate at the base incorporates three drain sumps and two recirculation sumps as well as the instrument tunnel and lift shaft sump, all with wall and floor plates. These areas are known as the wetted surfaces.

7.108. The liner plate forms the inner surface of the drain and recirculation sumps and the instrument tunnel. The instrument tunnel liner floor is protected by a concrete topping slab which is intended to prevent lateral buckling deformation of the liner due to thermal expansion. The liner plate within the sumps is not covered by any topping concrete.

7.109. These base liner sections are subject to general visual examination under the ASME XI requirements.

7.110. It is an additional requirement that a survey of Reactor Building internal structures for any effects of boric acid exposure is undertaken during each refuelling outage. The survey is completed in accordance with a station procedure and comprises a visual survey covering the areas of the containment annulus that are accessible, and those areas within the inner areas (loops) that are also accessible.

7.111. Boric acid can be circulated in the atmosphere of the Reactor Building by means of the Heating Ventilation and Cooling system in the event of primary coolant leakage.

7.112. Defects found during the course of the inspection are classified into the following categories:

> **1** – Emergency affecting safety-repair required immediately. Until final repair is completed, area is to be fenced off and a temporary repair to be effected.
>
> **2** – Not affecting safety – repair required to agreed timescale to prevent further deterioration.

3 – Not affecting safety – repair carried out to normal station maintenance programme, or other agreed timescale.

No – Indication or feature with no effect upon safety – no repair required.

(v) Interactions of liner and concrete containment (Equipment Hatch and penetrations)

7.113. The main liner and concrete containment includes large diameter penetrations that are included in the surveillance scope as specified in the Technical Surveillance Programme for each refuelling outage. The steel liner acts as a gas-tight membrane preventing release of radioactive material to the external environment. To ensure it performs this function, the liner was designed to remain within elastic limits under normal and exceptional loads. The liner is securely anchored to the pre-stressed concrete shell and the structural anchorage arrangement was designed to limit concrete strains and ensure elastic structural behaviour under all design loads. The anchors cannot be physically inspected and hence reliance is claimed on the inherent design margin and the presence of any secondary indications of distress from the detailed surface examinations.

7.114. The Equipment Access Hatch is a door fabricated from a carbon steel plate bounded by flange. The door which is internal to the reactor building forms a closure to penetration in the primary containment through which large items can be conveyed during outages. It contains 20 circumferential eye-bolts and two transverse seismic restraint eyebolts to hold the door in place.

7.115. During normal operation, the Equipment Access Hatch functions as part of the main liner membrane and is required to maintain leak tight integrity under normal operating and fault loading conditions.

7.116. The AMP requires inspection of the following seals:

- Secondary Containment – building gap seals.
- Interbuilding – flexible fire barrier seals.
- Special door seals for pressure and flood resistance.

7.117. The special door seals are inspected on a rolling programme by the Station civil maintenance contractor. Inspection results and any defects observed are recorded through the Work Management System. Any necessary maintenance work is carried out after Station approval to maintain them in a satisfactory condition.

7.118. Routine inspections of the large flexible secondary containment building gap seals and inter-building flexible fire barrier seals are performed in accordance with a station maintenance instruction. The results and any defects observed are recorded through the Work Management System. Any necessary maintenance work is carried out after Station approval.

7.119. Access to the interior of the PCCV is via two Personnel Access Airlocks forming part of the containment boundary that are inspected to confirm their continued structural integrity. Each Personal Access Airlock penetration consists of steel sleeve which penetrates the main PCCV sidewall. The sleeves are welded to the liner plate to provide containment boundary.

7.120. In addition to the large diameter penetrations there are 101 electrical and 124 mechanical penetrations through the liner to allow required services to pass from the reactor to other buildings. All of the penetrations are supported as they pass through the liner by thicker embedment plates welded into the liner. The welded joints between the penetrations and their embedment plates form part of the containment boundary. Under the Technical Surveillance Programme, these welds are inspected to ensure that they retain their structural integrity and do not compromise the containment leak tightness.

(vi) Seals (moisture barrier at barrel to basemat junction)

7.121. The junction between the liner wall and the concrete floor of the PCCV includes a layer of compressible polymeric material between the liner and the concrete. The overall design of the junction acts to reduce the risk of corrosion at the liner wall-floor joining weld and on any uncoated liner plate due to accumulation of moisture in that confined area. The floor and junction are painted with the same coating as used on the walls, leaving the top surface compressible filler obscured from view.

7.122. To address ageing degradation effects, the moisture barrier seal is subject to general visual examination around the full circumference to identify any defects. Acceptance criteria are based upon ASME XI Subsection IWE 3500.

Key design documents to inform the AMP

7.123. A Technical Specification for the main civil works was developed at the initial design stage (with dedicated chapters covering each of the structurally significant components). This document set down the design parameters and assumptions that underwrite the safety of the containment design strategy. In turn, this informed the scope of the key structural elements and acceptance criteria considered for surveillance and testing under the wider AMP remit.

7.124. As there was no specific UK design code applicable to concrete containment structures, a special document entitled “Design and Construction Rules for PWR Primary Containment” was established. These rules covered the design of the concrete structure and liner and were basically derived from the US Code ASME III Division 2 (Subsection CC) comprising three parts:

- Part 1 covers the definition, loads to be considered, loading combinations and allowable design limits.
- Part 2 is the technical specification for main structural components viz concrete, pre-stress and reinforcement.
- Part 3 shows how the ASME code has been adapted to provide a UK equivalent.

7.125. The principal differences between the ASME and UK design and construction rules are:

(a) The definitions are more in line with the UK pre-stressed concrete pressure vessel code BS4975.

(b) In line with BS4975, an ultimate load requirement is specified for the containment.

(c) Design allowable values are expressed in UK terms e.g. concrete as 28-day cube strength and reinforcement in yield for UK steel supply.

(d) For the liner, guidance is given for assessment of anchor failure taking account of non-linear behaviour under temperature loading.

(e) A limit is placed on liner tensile strain and advice is provided for addressing peak strains in localised areas.

7.126. As in the case of other civil structures, the design of the containment structures has been considered under the remit of the PSR process. These reviews have concluded that the original design basis remain adequate. Furthermore, as noted above, separate Discipline Reviews have been carried out, which considered revisions to codes and standards, and this concluded that there were no significant changes to the requirements of the ASME code.

7.127. Four main classes of concrete design were used for the PWR concrete structures:

- C5 - A mass fill concrete incorporating 50% Pulverised Fly Ash (PFA).
- C1 - A structural concrete incorporating 40% PFA.
- C2 - A structural concrete incorporating 40% PFA.
- C11LW - A lightweight concrete incorporating sintered PFA lightweight aggregate.

7.128. The C5 mix was used beneath the foundations of higher structures where buildings with differing foundation levels were constructed adjacent to each other. Class C1 concrete was used in the primary containment and the secondary internal structures of the reactor building.

7.129. Concrete of class C2 was used in the construction of the other power station buildings and the lightweight concrete class C11LW was used for the secondary containment dome.

7.130. Details of the various material properties for the concretes used in the construction of the PCCV are retained in the station lifetime records.

7.131. For the PWR primary concrete containment, un-grouted tendons of the PSC Freyssinet K Series type were used to permit withdrawal and in-service inspection and re-stressing of tendons should this be required. Corrosion protection of the un-grouted tendons was provided by a factory applied corrosion protection system applied to the pre-stressing strand. Following installation of the tendons the ducts were filled with a corrosion inhibiting soft wax.

7.132. There are 107 horizontal pre-stressing tendons, each spanning 240° of circumference, and 74 vertical (inverted U-shaped) tendons passing from the base and over the dome, forming an orthogonal grid on the dome. Each tendon consists of 37 No. 15.2 mm diameter 7-wire compacted stabilised strands manufactured to BS5896:1980 and the requirements of the PWR Civil Works specification.

Key standards and guidance to inform the AMP

7.133. The IAEA Safety Guide NS-G-2.12 Ageing Management for Nuclear Power Plants sets down the principles for managing nuclear safety structures to ensure the availability of required safety functions throughout the service life of the plant, with account taken of changes that occur with time and use. This requires addressing both physical ageing of structures, systems and components (SSCs), resulting in degradation of their performance characteristics, and obsolescence , i.e. their becoming out of date in comparison with current knowledge, standards and regulations, and technology.

7.134. A foundation for effective ageing management is that ageing is properly taken into account at each stage of a plant's lifetime, i.e. in design, construction, commissioning, operation (including long term operation and extended shutdown) and decommissioning. The effective management of the containment structures ageing is a key element of the safe and reliable operation of nuclear power plants.

7.135. The primary objective of the Safety Guide NG-S-G-2.12 is to provide recommendations for managing ageing of SSCs important to safety in nuclear power plants, including recommendations on key elements of effective ageing management. As a nuclear licensee operator EDF-NG comply with the principles which support the establishing, implementing and improving of the systematic ageing management programmes for the containment structures. The EDF-NG approach aligns with the Safety Guide and seeks to focus on managing the physical ageing of the containment structures and components important to safety. The Guide also provides recommendations on the safety aspects of managing obsolescence and on the application of ageing management for long term operation.

7.136. IAEA also produce a programme of International Generic Ageing Lessons Learned (IGALL) reports for Nuclear Power Plants. These reports supplement the IAEA Safety Reports Series No. 82, Ageing Management for Nuclear Power Plants. For structures, systems and components important for safety, the following information is presented:

- a generic sample of ageing management review tables;
- a collection of proven ageing management programmes; and,
- a collection of typical time limited ageing analyses.

7.137. The IGALL documents relevant to the ageing management of the PWR containment structures are listed below:

- AMP 301 ISI for Containment Steel Elements.
- AMP 302 ISI for Concrete Containment.
- AMP 304 Containment Leak Rate Testing.
- AMP 308 Protective Coating.

7.138. These safety reports describe practices and techniques for the inspection, mitigation of ageing degradation, corrective action including repair methods, and operating experience for concrete containments. They also provide general guidance for developing an effective ageing management process for concrete containments based upon the following principles.

- Understanding ageing mechanisms.
- Detection of ageing effects.
- Monitoring and trending of ageing effects.
- Mitigating ageing effects.
- Acceptance criteria.
- Corrective actions.

7.139. The IAEA guidance provides specific information for in-service inspection (ISI) for ageing managing of containment reinforced concrete including un-bonded post-tensioning systems. It is judged that the EDF-NG procedures for ageing management align with the intent of the IAEA guidance and there are no significant gaps identified.

7.140. The following ASME boiler & pressure vessel codes are used to develop the in-service inspections:

- ASME Code, Section XI, “Rules for In-Service Inspection of Nuclear Power Plant Components”, Subsection IWA “General Requirements”.
- ASME Code, Section XI, “Rules for In-Service Inspection of Nuclear Power Plant Components”, Subsection IWE “Requirements for Class MC and

metallic liners of Class CC components of light-water cooled plants", 2001 edition.

- ASME Code, Section XI, "Rules for In-Service Inspection of Nuclear Power Plant Components", Subsection IWL "Requirements for Class CC concrete components of light-water cooled plants".

Research and Development

7.141. Historically, a number of research projects have been commissioned. Many of these initiatives were primarily relevant to AGR PCPVs but the studies, where applicable, have been read across for interpretation to the PWR containment structures.

7.142. The general areas of study that have been considered in these research projects are summarised below:

- Effects of elevated temperatures: Assessment of structural integrity of concrete at elevated temperatures.
- Concrete properties: General in-situ properties of concrete in safety related structures. Effects of cycles of elevated temperatures/cooling on restrained concrete. Ageing/durability at elevated temperatures.
- Behaviour of concrete at elevated temperatures, including multi-axial loading and development of constitutive models.
- Fracture energy at elevated temperatures.
- Survey methods: Review of advanced survey techniques and their application.
- Tendon corrosion/protection: Long-term performance of greases used for tendon protection. Extent of corrosion in unprotected tendons.
- Tendon loads: Measurement of tendon loads. Tendon load profile modelling. Instrumentation for monitoring of tendon loads.

7.143. The research studies to date have broadly supported and enhanced understanding of a wide range of potential ageing and degradation issues. In particular, the investigations into the performance of grease for protection of tendons, and the extent of corrosion in unprotected tendons, provided informative results. These results are of practical use in justifying the long term integrity of pre-stressing tendons. The general conclusions to date confirm that the protection provided by the grease remains effective.

7.144. There are a number of projects that have considered the structural integrity of concrete at elevated temperatures, together with the properties of concrete at both normal ambient and elevated temperatures. Some of these projects have also considered ageing and durability issues related to vessel concretes. The findings of these studies have been incorporated into finite element analysis work, and structural integrity assessments, that have been performed for the PCCV under fault conditions.

7.145. Finally, it is noted that the multi-axial testing of concrete at elevated temperatures is still ongoing, and it is anticipated that the results of these studies will provide data for the further refinement of concrete constitutive models in future analysis work.

Operational experience (OPEX)

7.146. As a fundamental part of AMP, operational experience and feedback is routinely reviewed to ensure that any issues of relevance to the PWR containment structures are assessed and to allow any lessons learned to be incorporated into the ageing management strategy. The following key issues are highlighted:

Internal OPEX

7.147. There are two cases of performance related events that relate to the containment structure.

- Failure of relief valve to close following Trevi-testing leading to loss of coolant.
- Leakage from the RPV head O-ring seals.

7.148. The events were reported at INES Level 1. In both cases, some flooding exposed the floor concrete, liner, wall-base moisture barrier and some plant fixings to boric acid solution. It is known that boric acid is very corrosive to steel structures and components with the potential to "reactivate" at a later stage as boric acid crystals rehydrate. On neither occasion was immediate damage noted but the requirement to maintain an inspection regime throughout service for the affected areas was recognised.

7.149. The AMP was enhanced in recognition of these events. Formal procedures were developed to include regular inspections of the Reactor Building internal structures to detect any instances of damage due to boric acid exposure. Leak detection systems were also improved.

External OPEX

(i) Sizewell B Farley tendon failure

7.150. An incident occurred in May 2012 at Unit 1 of the Joseph M. Farley Nuclear Plant, which involved the failure of a post-tensioned hoop tendon in the containment building. The failed tendon ejected into an adjacent stairwell, together with the tendon anchorage, shims, anchorage cover and a quantity of grease. It was established that the cause of the incident was the failure of the tendon anchorage head at one end of the tendon, caused by hydrogen stress cracking. This occurred, despite the anchorage being protected from the surrounding environment by grease. Further analysis of the grease from the tendon, however, showed that oxidation and degradation of the grease had occurred.

7.151. It was noted that the grease at Farley contained an additive of over-based calcium sulphonate, together with calcium hydroxide, and that breakdown of the grease could release hydrogen to potentially cause hydrogen stress cracking. This could also combine with hydrogen released due to the presence of moisture that was detected in the grease.

7.152. These conditions are unlikely to occur on the Sizewell B containment because the external surface of the containment is protected from the external conditions either by adjacent buildings, or the secondary containment enclosure building, and there is no cooling system to introduce water into the environment.

7.153. In addition the current inspection regime at the Sizewell B containment includes activities which are required to detect any signs of ingress, or potential ingress, of water into the pre-stressing system. These include:

- The analysis of samples of corrosion protection medium (grease) and inspection for the presence of any free water.

- Routine visual checks of the containment surface and tendon anchorages for any evidence of wetting or corrosion, together with inspection of the secondary containment enclosure building to confirm continuing weather tightness.

7.154. In addition to the above requirements, the five-yearly inspection regime includes a detailed visual examination of nominated tendon anchorage hardware and surrounding concrete. It can therefore be concluded that, unlike the PCPVs at the AGR stations in the EDF-NG fleet, where water can be found in the tendon end covers and in the ducts due to PVCW leaks, the presence of water in the Sizewell B PWR containment pre-stressing system is very unlikely, and would be detected at an early stage if this was to occur.

(ii) <u>Davis Besse shield building cracks</u>

7.155. The Davis-Besse containment vessel and shield building lacked an access opening of sufficient size to permit removal of the old reactor vessel head and installation of a replacement vessel head. Therefore, during the 17-mid-cycle outage, the licensee cut a temporary access opening in both structures of sufficient size to support the head replacement.

7.156. During construction of the access opening subsurface cracking located near the outer reinforcement mat was discovered. The cracks were discovered in October 2011, and consisted of subsurface laminar cracking.

7.157. Root cause assessment suggested that this issue was associated with extreme environmental conditions of moisture penetration and freezing.

7.158. The Davis-Besse shield building is a reinforced concrete structure with no pre-stressing system, and the delamination is essentially attributed to environmental factors associated with exposure to the external environment.

7.159. The external surface of the Sizewell B containment is protected from the external conditions either by the secondary containment structure or the adjacent Auxiliary Building. It was concluded that this is not a potential issue for the main containment structure.

7.160. Similarly, the reinforced concrete secondary containment structure is covered by external cladding, and is therefore protected itself from the external conditions. It can therefore be concluded that this is not an issue for the Sizewell B PWR containment.

(iii) <u>Temelin 1 tendon failure</u>

7.161. A tendon failure occurred at Temelin 1 containment building in the Czech Republic in April 2007. The failure occurred in a dome tendon during start up after a refuelling outage. The event report noted that there had been several tendon failures on this plant. All previous failures had occurred during installation or initial stressing, or during load testing. These failures were attributed to overstressing caused by a number of factors, but generally exacerbated by deformation at the location of eye rings during installation and stressing. Proposed corrective actions included increasing the size of the tendons (number of wires per tendon) and modification to equipment and procedures for the manufacture, installation and testing of the tendons.

7.162. It is noted in the event report that a number of tendon failures had occurred at the plant, and that these are attributed to a number of deficiencies in the design, manufacture, installation and stressing of the tendons. These deficiencies resulted in overstressing, yield, and ultimately failure of a number of tendons. There are no indications that similar problems are associated with the Sizewell B PWR containment tendons.

(iv) South Ukraine Unit 3 tendon failure

7.163. A tendon failure occurred in the South Ukraine Unit 3 containment building in Ukraine on 28 January 2001. The failed tendon was withdrawn, and an investigation showed that the failure was initiated by the rupture of wires as a result of corrosion. The corrosion was attributed to 'loss of lubrication during the process of tendon installation', i.e. a lack of corrosion protection medium on the tendon wires. The report also noted that 'irregular distribution of tension force between interior and outer tendon wire layers contributed to the wire damage'.

7.164. The report for this incident clearly notes that the failure is attributed to corrosion as a result of poor corrosion protection. The ISI regime for the Sizewell B PWR containment includes tests and inspections that are intended to avoid the occurrence of issues of this nature. These procedures are compliant with ASME requirements, and are subject to regular review.

(v) Seabrook Alkali-Silica Reaction (ASR)

7.165. The Safety Evaluation Report issued by the NRC in June 2012, noted 'Open Items Related to the License Renewal of Seabrook Station'. One of these open items relates to ingress of groundwater, and the effects of ASR on below grade concrete structures. In particular, it is noted that cracks due to ASR have been observed in different Seabrook plant concrete structures, including the concrete enclosure building.

7.166. There was therefore concern that the groundwater may have penetrated the concrete containment wall and come into contact with the containment liner plate. The report notes that this could result in through-wall corrosion of the containment liner plate.

7.167. ASR is a general potential issue for all of concrete civil structures. However, the aggregates used for the Sizewell B PWR containment structures were chosen such that they were not vulnerable to ASR effects and no visible evidence has been observed to date.

7.168. The ingress of groundwater to the PWR lower liner is recognised by EDF-NG as a potential issue to be monitored going forward and currently enhanced borehole sampling and testing takes place to assess the ground water chemistry. The AMP would benefit from wider utility feedback on this issue to assess its longer term credibility and risk to the containment and any enhanced monitoring or preventative measures that could be investigated (see paragraph7.306).

7.1.2.2 Hinkley Point C (NNB)

7.169. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

7.170. This section describes the contingencies taken at the design stage of the EPR^{TM} UK HPC to address the ageing mechanisms.

7.171. Based on the applicable codes and standards, on the OPEX and on the specific conditions affecting the EPR^{TM} UK HPC, this section list the requirements specified for the selection, qualification and supply of the NAR examples listed below.

(i) Concrete structures

7.172. Concrete are designed in compliance with the specifications of BS EN 206 and BS EN 8500-1 and -2. The type of concrete (strength class, target consistency, Dmax, grading, water content, type and content of cement) is adapted to the environment (exposure classes of BS EN 206) in which it is placed with an intended working life of at least 100 years.

7.173. The exposure class for the EPR^{TM} UK HPC Inner Containment is XS1 and the exposure class for the EPR^{TM} UK HPC Airplane crash shell, outer containment and common raft is XS3. The provisions in force in the relevant standards and recommendations regarding concrete durability are implemented as presented below.

7.174. An extensive testing programme has been undertaken to demonstrate the compliance of EPR^{TM} UK HPC concrete mixes with the criteria presented here below. It shall be noted that these tests support ageing management as:

- Non reactivity tests and modelling of temperature distribution in large pours mitigate the risk of delayed ettringite formation support the durability of the concrete
- Creep tests of pre-stressed concrete allow to control pre-stressing losses
- Air permeability and porosity tests of the APC and Outer containment concrete allow to control the kinetics of concrete degradation

7.175. The content of chlorides Cl- contributed by all concrete constituent materials complies with BS EN 206:

- class Cl 0.20 (max 0.2% of Cl- by mass of cement) for reinforced concrete,
- class Cl 0.40 (max 0.4% of Cl- by mass of cement) for reinforced concrete with a CEM-III cement or a C-III combination,
- class Cl 0.10 (max 0.1% of Cl- by mass of cement) for pre-stressed concretes (inner containment wall of the Reactor Building).

7.176. The chloride content of concretes in contact with stainless steel liners or any other stainless steel elements where there is a risk of leak is limited to 150g/m3.

7.177. The sulphide content S2- contributed by all concrete constituent materials is limited to:

- 0.7% of the mass of cement or combination for pre-stressed concrete in the case of post-tensioning;
- 0,5% of the mass of cement or combination for pre-stressed concrete in the case of pre-tensioning.

7.178. These requirements aim to limit the appearance of ageing mechanism such as corrosion.

7.179. The risk of damaging alkali-silica reaction is minimized in accordance with the guidance set out in BRE Digest 330.

7.180. The qualification test includes a non-reactivity test for the nominal concrete mix. The acceptance criteria are:

- In the case of a concrete mix containing only CEM I, the expansion at five months shall be less than 0.02%
- For all other cements in the concrete mix, the expansion at nine months shall be less than 0.02% or less than 0.03% at one year

7.181. Special requirements for large volume of concretes are in force in order to avoid the risks of delayed ettringite formation and thermal cracking.

7.182. The approach for preventing the risk of Delayed Ettringite Formation is based on the LCPC guide Recommendations for prevention of disorders due to delayed ettringite formation.

During construction on site, the maximum temperature reached in the concrete shall be less than 65^{o}C.

7.183. Regarding the control of thermal cracking, the temperature difference between the central point of the concrete structure and its surface (measurement point situated at the concrete cover level), during the heating and cooling phases of the concrete, shall comply with the following conditions:

- The difference shall be less than or equal to 25^{o}C in reinforced concrete
- The difference shall be less than or equal to 20^{o}C for unreinforced concrete

7.184. The Hinkley Point site presents aggressive ground conditions exposing structures to sulphate attack. In accordance with the BRE Special Digest 1:2005, the structure is designed to be protected against chemical attacks caused by the ground.

7.185. In order to ensure the common raft concrete endure the aggressive ground conditions, the design class is set to DC4. In addition, an extra layer of 10cm of DC4 concrete is added as an additive protective measure to comply with the 100-year structure lifetime.

7.186. For pre-stressed concrete the qualification test also includes:

- A test to determine the coefficient of thermal expansion
- A creep test

7.187. For outer containment concrete the qualification test also includes:

- An air permeability test for hardening concrete in accordance with XP P 18-463 standard
- A test to determine the porosity. The permeability of the concrete dried at 105^{o}C < $1x10^{-16}$ m/s

(ii) Steel reinforcement

7.188. Steel for the reinforcement of concrete, in the form of bars or coils, is certificated by UK CARES as complying with BS 8666. Steel fabrics are certificated by UK CARES as complying with BS 4483.

7.189. Steel reinforcement is grade B500C with a specified value of yield strength Re,act/Re,nom ≤ 1.3.

7.190. Indented steels are not permitted except for specific prefabricated elements with prior approval of NNB.

7.191. The suitability of steels intended to be re-straightened on site, is mentioned in the conformity certificate. The use of reinforcement bars with diameters strictly greater than 40 mm is submitted to the Employer's approval.

7.192. For steel fabrics, the nominal diameter shall be less than 16 mm.

7.193. Steel reinforcement protection against ageing is mainly provided by the concrete cover. The concrete cover of the reinforcement is at least equal to the minimum concrete cover as defined in EN1992-1-1.

7.194. The cover requirements apply to all reinforcement materials, whether or not they are the principal reinforcement bars or secondary fixings.

(iii) Pre-stressing system

7.195. Pre-stressed concrete systems are subject to a European Technical Approval (ETA) and have a certificate of conformity to the ETA and a CE mark declaration of conformity.

7.196. The CE mark and ETA are issued by independent certification organisations accredited by UKAS (or equivalent to UKAS) and the organisation issuing the ETA is also authorised by the European Organisation for Technical Approvals.

7.197. Further to the demonstration carried out in the scope of technical approval and the scope of certification, the contractor proves that the pre-stressing system is able to be used for containment of nuclear power plant.

7.198. This demonstration is based on the experience feedback of the construction of nuclear containment using adherent pre-stressing system and takes into account:

- Material and placing of the ducts in order to control friction coefficient inside the ducts
- Installation of strands, tensioning and jack specification in order to control possible deviation between tension of the strands
- Grouting with cement grouts to assure a maximum filling of ducts and maximum protection against corrosion

(iv) Pre-stressing seals

7.199. The pre-stressing units are composed of 1860MPa minimum ultimate tensile strength strands. From ageing perspective, strands shall comply with the requirements listed below:

- Strands shall be constituted of special non-alloyed carbon steels in accordance with BS EN 10020
- Strands shall comply with Pr EN 10138-3 and especially:
    - The relaxation of strands shall be evaluated at 20^{o}C with loads equal to 70% and 80% of the nominal force and the results shall be respectively less than or equal to 2.5% and 4.5% at 1000h. The relaxation test shall be performed for a period of at least 240h and may be extrapolated to 1000h.
    - The relaxation of the strands shall be evaluated at 40^{o}C under a load equal to 70% of the nominal force and the results shall be less than or equal to 3% at 1000h. the relaxation test shall be performed for a period of at least 240h and me be extrapolated to 1000h
    - Corrosion resistance of strands under stress solicitation shall be evaluated according to BS EN ISO 15630-3 using aqueous solution A. The results shall be considered compliant when all the durations are greater or equal to 1.5h and when the average duration is greater than 4h.

7.200. Anchorage components or parts of components not embedded in the concrete (exposed to atmospheric corrosion after formwork removal) shall be protected with a coating system of type PED200 category I.The coating system shall be corrosion proof in a highly corrosive atmosphere (category C5-M as per BS EN ISO 12944-2).

7.201. All necessary precautions are taken to protect the coating system during transport, storage and installation, up to the point where the installation is completed.

7.202. The two broad families of products can be used for permanent protection of pre-stressing strands:

- Cementitious grout
- Flexible products for instrumented tendons

7.203. Regarding the cement grout, no element is permitted in the admixtures or additions which may give rise to corrosion of steel.

7.204. Grout shall be adapted to the geometry and profile of the ducts in order to ensure complete filling. The filling capacity for ducts is checked during the qualification phase for the injection technique. To justify that the grouting methods are suitable a series of 1:1 scale mock-up tests on each type of tendon (horizontal (slightly and highly deviated), vertical and gamma dome section) are planned to qualify the injection technique, grout products, personnel and equipment to be used and to demonstrated that correct filling of the pre-stressing ducts can be achieved.

7.205. Correct filling of the ducts is necessary to ensure protection of the strands against corrosion, confirm the quality of bonding between grout and strand and then confirm the uniformity of sections for correct transfer of internal stresses.

7.206. An initial successful trial is required to validate grout characteristics and injection techniques, followed by two additional trials to confirm the satisfactory result.

7.207. Operational experience on previous mock-ups is presented below:

- Civaux nuclear power plants (France, geometrically representative of EPR™ ducts): three tests were carried out on purely vertical tendons using retarded grout – result: “perfectly filled conduits and upper end-caps”. Three tests were carried out on pure horizontal tendons, using retarded-thixotropic grout: good filling (criteria met). Three tests were carried out on highly deviated horizontal tendons, using retarded-thixotropic grout: These tests were not successful, with multiple air voids being found in deviated sections. Therefore, the injection technique was changed to vacuum injection. With the vacuum injection method, good filling of the conduits was achieved.
- Olkiluoto and Flamanville EPR™: successful mock-ups.

7.208. Cement is of type CEM I. From an ageing management perspective, cement for the grouting of conduits and anchorages for prestressed cables satisfy the following requirements:

- Chloride content Cl- < 0.05% by mass of cement,
- Sulphur content S2- < 0.01% by mass of cement,
- Free of any other elements likely to lead to corrosion of the reinforcement

7.209. In addition to the quality and conformity control tests required by the standards in force, the Contractor carry out a corrosion test under tension in accordance with BS EN ISO 15630-3, on the batch of pre-stressing steel at the beginning, middle and end of the strand installation period. The test is carried out using aqueous solution A under a force equal to 80% of the mean value of the maximum force.

7.210. The result of the test shall comply with the following values:

- 1.5h of individual values
- 4h for the median lifetime value as defined by BS EN ISO 15360-3

7.211. Batches or bundles of steel subjected to testing are not used in the works before the results of the tests are known and are compliant.

7.212. The following quality control measures are implemented at the production facility (at the point of departure of the materials) and at the point of delivery, for each batch:

- Preliminary protection: oiling of the steel at the production facility with an oil emulsion and packing suitable for the transport,
- Labelling: the labelling shall assure complete traceability of the delivered product,
- Each coil delivered to the site shall be subject to a visual inspection to check the geometry of the bundles and the strand surface condition. Strands showing any signs of oxidation, which remain after cleaning with a rag, or which do not satisfy the criteria A or B of the corrosion scale described in the table below are rejected:

Criteria	Visual characteristics	Remedial measures
A	- uniform colour - light staining from localised oxidation - few scratches on the bare metal (brilliance) - no foreign material - no indentations	No cleaning required for the inspection
B	- some staining - general oxidation but light in nature - traces of scratches weakly visible on the bare metal - no indentations - some detachment of the oxide film	The surface shall be cleaned with a rag before the metal is visually observed

7.213. Quality control checks of steel tubes are performed to ensure that the following requirements are satisfied:

- The ends of the tube shall be deburred,
- The required surface treatment shall be in place,
- Tube dimensions shall be in accordance with the relevant standards,
- Level of oxidation,
- Tubes shall be clean (internally and externally),
- Tubes shall be free of blemishes, cracks and local deformations.

7.214. Quality control checks of sheaths are performed to ensure that the following requirements are satisfied:

- Sheaths shall be free of any internal and external corrosion at the point of delivery,
- The ends of the sheaths shall be deburred,

- Sheaths shall be clean and free of any deformations and signs of impact,
- The geometry of the sheaths (internal diameter, height of profile and pitch) shall be suitable for their application,
- Sheaths shall be CE marked.

(v) <u>Metallic containment liner (with anchorage system), external containment penetration sleeves, internal containment penetration sleeves, anchorage of the various equipment supported by the containment wall [e]</u>

7.215. With the exception of S235JR grade products, all products are delivered at least with a Type 3.1 inspection certificate in compliance with standard BS EN 10204.

7.216. S235JR grade products are delivered with a Type 2.2 inspection certificate in accordance with BS EN 10204 with the condition that tensile tests and chemical analysis are statistically performed to verify the BS EN 10025-2 requirements.

7.217. From an ageing perspective, requirements are set to mitigate corrosion kinetics.

7.218. Plates for containment parts (including liner, anchor plates, sleeves, blind flanges and anchorage rings) are grade P265GH or higher in accordance with the requirements of BS EN 10028-3 with a specified impact testing at -20d C.

7.219. Structural sections (including stiffener and other anchor elements) are grade S235JR, supplied in accordance with BS EN 10025-2.

7.220. Inner containment penetration sleeves are made of P265GH according to BS 10216-2 or A106GrB according to the ASTM standard.

7.221. Outer containment sleeves are made of S235, S275 or S355 steel according to standards BS EN 10219-1 or BS EN 10025.

7.222. Studs for containment liner (diameter 8mm): Concerning the chemical composition, studs are made of grade S235J2 material in accordance with BS EN 10025-2.

7.223. Studs for anchor plates (diameter >8mm): studs are made of grade S235J2 or higher in accordance with BS EN 10025-2.

7.224. In order to protect the liner against corrosion, the entire liner containment surface not in contact with concrete is painted.

7.225. During storage, and in general if there is a risk of persisting contact with an aggressive atmosphere, all faces concerned are protected against corrosion risk.

(vi) Sealing of penetrations [g]

7.226. Caulking filling in the reactor building penetrations will retain mechanical properties for a minimum of ten years and will not be subject to dimensional variation over time, leading to shrinkage, cracking or becoming separated from the walls or from any of the various penetrating items.

7.227. The penetration sealing processes will be repairable with materials that are compatible with original product.

7.1.3 Monitoring, testing, sampling and inspection activities for concrete structures

7.1.3.1 Operating reactors (EDF-NG)

7.228. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

7.229. In-Service Inspection (ISI) of the primary containment is controlled by a Technical Surveillance Programme that is developed prior to each refuelling outage and which determines the surveillance scope. A refuelling and maintenance outage is undertaken every 18 months.

(i) <u>Concrete structures</u>

7.230. The original containment construction was confirmed by a pre-service Structural Overpressure Test (SOT), when the structure was pressurised to a gauge pressure of 1.15 times the (gauge) design pressure, 0.397 MPa(g) = 0.498 MPa(a). The leak-tightness of the containment was demonstrated by a pre-service Integrated Leak Rate Test (ILRT), and is demonstrated by continuing in-service ILRTs and Local Leak Rate Tests (LLRTs) on the containment penetrations.

7.231. The ILRT involves measurement of the leakage from the Primary Containment at the postulated accident pressure i.e. 0.274 MPa(g) = 0.375 MPa(a). The current interval for this test is 10 years. Surface examinations are conducted both before and after the ILRT to assess for any changes in structural condition.

7.232. The LLRTs test the leak-tightness of Containment Isolation Valves (CIV) and pressure boundary seals at the accident pressure of 0.375 MPa(a) during each refuelling outage.

7.233. The surfaces are visually inspected (noting that the inner face of the containment is steel lined including sumps and drains) including the use of roped access techniques to ensure adequate coverage and to allow inspectors to examine all non-obstructed parts of the surface. The examination of the containment includes:

- Containment surfaces – main liner (plate surface, welds), Equipment Access Hatch, Personnel Access Airlocks (plate surface and attachment welds, structural welds, seals, bolts), pipework and electrical penetrations (plate surface, welds, bolts).
- Containment surfaces – wetted surfaces (Instrument Tunnel liner, Recirculation Sumps, Floor Drain Sumps).
- Containment surfaces – moisture barrier at liner/floor junction.
- Containment external concrete surface.
- Visual inspection of accessible concrete surface areas, together with suspect areas as defined in the relevant civil ISI plan.
- Examination of the polar crane corbel to check for degradation.
- Review of the embedded instrumentation.
- Local Leak Rate Testing.

7.234. A full report of the ASME XI inspections and the associated containment inspections is provided in an Appointed Examiner report that presents the findings from each refuelling outage cycle and which is issued within 90 days of the completion of the outage.

7.235. The main findings to date conclude that the primary containment appears in generally good condition. There have been a number of indications noted of fine cracking in several areas of the structure, although these appear to be of a similar scale to those observed during previous inspections, and consistent with that expected due to restrained thermal cracking from construction and further drying shrinkage since construction.

7.236. Indications of fine cracks in the concrete have been observed around the main penetrations in the Steam and Feed Cells, some crossing the ligament between the steam pipe penetrations. At least one of these cracks was initially observed during the first in-service inspection of the concrete. It was judged that these fine cracks are likely to have been caused by restrained thermal cracking during construction and drying shrinkage in the concrete surface layers and have existed since construction, covered by the paint coating on the concrete. An assessment has concluded that there is no structural significance to this cracking. Based upon APEX advice, a sample of three of the cracks have demountable mechanical strain gauge points fitted to allow monitoring, to determine if there is any significant change in their width over time.

7.237. Some minor defects are present in the concrete and judged to be related to issues from construction, such as poor compaction, quality control on formwork, and shrinkage cracking. No defects have been observed of a significance that would have an adverse effect on the overall structural integrity of the containment. It is also noted that the PCCV is sheltered (by the Secondary Containment and the Auxiliary Building), and therefore not subjected to the direct effects of exposure to the weather. Therefore, it is considered unlikely that any significant deterioration in the longer term will occur.

7.238. A summary of the concrete surface defects follows:

- Many of the very fine cracks were noted in earlier inspections and no longer detectable, having since closed, or been painted.
- A small number of newer cracks continue to be observed, but these are fine cracks, and of no structural significance.
- The features noted to date do not represent any degradation of the condition of the PCCV, and do not raise any concerns for its continuing integrity and fitness for service.
- The type and number of indications and defects that continue to be observed are similar to those that have been observed during previous inspections. This indicates that no significant change or degradation in the condition of the concrete of the PCCV structure is taking place.
- Readings from the demountable mechanical strain gauge points fitted on three fine cracks at the Steam & Feed penetrations indicate that the cracks are small (approximately 0.15 mm) and their behaviour is as expected. Hence, the fitting of demountable mechanical strain gauge points may be considered as a prudent measure of the AMP to monitor longer term crack behaviour in this area.

7.239. A visual examination of the secondary containment to comply with Surveillance Requirement of the Technical Surveillance Programme is undertaken routinely during the fuel cycle period. This includes confirmation of continuing weather tightness. Visual checks have been undertaken at 6-monthly intervals but due to the absence of any adverse indications, they are now performed once per operating cycle. This change to the in-service inspection regime was justified as per the safety case modifications process.

7.240. The purpose of these examinations is to confirm that there are no conditions which could adversely affect the condition of the PCCV pre-stressing system and lead to corrosion of pre-stressing tendons in ducts which may not be completely filled with corrosion prevention material.

7.241. The accessible surfaces of the remaining concrete structure that comprise the containment boundary (including Auxiliary Building) are inspected at six-monthly intervals to check for water ingress. No adverse indications have been observed.

7.242. Instrument data logging and analysis is not a mandatory part of the Technical Surveillance Programme for assessing the on-going health of the concrete containment structures. However, under the wider remit of the AMP it is carried out as a supplementary measure to further underwrite structural integrity. The embedded instrumentation of primary interest comprises vibrating wire strain gauges, VWSGs for the measurement of strain & resistance foil strain gauges, RFSGs for measurement of concrete temperature.

7.243. Overall the data indicates the expected strain behaviour of gradually increasing compressive strain in the primary containment structure due to concrete creep. There have been no indications of the structural behaviour deviating from the expected design condition.

7.244. Structural behaviour based upon the data from the most recent ILRT indicates that the primary concrete containment continues to respond under pressure cycling in a similar manner to that observed during the original Structural Overpressure Test (SOT), pre-operational ILRT and previous in-service ILRT. The structure continues to exhibit elastic deflections of the expected magnitude and sense, with measured strain values returning to normal operational values in each case.

7.245. There is no operational requirement for the embedded instrumentation to remain in a functioning condition since the original SOT. However, there is potential value from the data obtained to inform and support ageing management in future. The instrumentation recording system has been afflicted by a number of reliability issues since the SOT and this has led to limited data availability for some periods in addition to general gauge obsolescence. As instruments tend to fail over time, this situation is unlikely to remain at its current level of informative use.

7.246. In terms of the long term ageing of embedded instrumentation the AMP would benefit from other utility feedback and best practice sharing on the retrospective installation of concrete performance monitoring systems to address obsolescence issues. This would include management practices for maintaining, refurbishing and replacing of concrete strain measuring devices to support long life surveillance, taking cognisance of improvements in technology and remote monitoring (see paragraph 7.306).

(ii) <u>Steel reinforcement</u>

7.247. The on-going condition of the steel reinforcement in both the primary and secondary concrete containment structures & Auxiliary Building is informed by the detailed concrete surface examinations for any evidence of underlying corrosion. No evidence to date of reinforcement degradation has been found.

(iii) <u>Pre-stressing systems</u>

7.248. The examinations consist of four main activities, as follows:

- Examination of tendon force and elongation measurement for a sample of tendons.
- Tendon wire and strand examination and testing for a sample of tendons.
- Detailed visual examination of all tendon anchorages and surrounding concrete.
- Analysis of samples of corrosion protection medium and any free water.

7.249. The pre-stressing system performance and component integrity are routinely monitored. Tendon load levels at the anchorage are monitored by the lift-off technique. The load in one vertical and one horizontal tendon (the 'common tendons') is measured at each surveillance; the others are randomly selected from the population not already tested. One tendon of each type is de-tensioned (not a 'common tendon') and a strand withdrawn and replaced for close visual examination to detect any evidence of degradation of the strand material.

7.250. The anchorage areas are visually inspected to detect signs of damage to the anchorage, wedges and strand ends. Any free water in the duct is chemically analysed to determine its potential to cause corrosion. Samples of corrosion protection material are taken from the tendon surface, and analysed to confirm its continuing efficacy.

7.251. The pre-stress loads that were used in the design made allowance for longer term losses due to friction along the length of the tendon. It is well known that further losses will occur due to relaxation of the pre-stress loading during the life of the PCCV. These may result from a combination of elastic shortening, creep, shrinkage and temperature of the concrete, together with relaxation of the steel. The potential issue is whether the tendon load levels may reduce to levels below the minimum allowable values.

7.252. The in-service inspection requirements include tendon load checks which are used to monitor the variation in tendon loads during operational life, including the development of trends, and hence predict future load variation.

7.253. The status of the tendon loads is summarised below:

- The results of the load checking operations to date show that, in general, and as expected, the tendon anchorage loads are slowly reducing.
- The available results have been filtered to remove any results that can be identified as being unrealistically out of the expected range, probably caused by systematic errors in measurement. The remaining results are analysed to provide predictions of load trends, and the times to reach minimum load values is determined for the differing group of tendons – i.e. hoop or vertical.
- In the case of the hoop tendons, a logarithmic regression is used to derive the average of the derived lift off loads (LOLs) for all hoop tendons from the filtered results. This shows that the predicted trend line is flattening out and the prediction for reaching the average required lift-off load, for all hoop tendons, is well beyond the operational life of the station.
- In the case of the vertical tendons, based upon the most pessimistic analysis it is shown that anchorage loads in the vertical tendons may reduce below the minimum requirements by circa 2045, which is beyond the current operational life of the station. This is considered to be an unlikely prediction, and it is more than likely that the observed rates of load loss will reduce, such that anchorage loads remain acceptably above the minimum requirements.

7.254. Samples of the corrosion protection medium are taken from the anchorage region of the tendons which are de-tensioned, as required by an approved station procedure. In addition, a control sample of unused Visconorust 2090-P4 corrosion protection material is used for benchmarking.

7.255. The analyses to date show results in compliance with the acceptance criteria (see Table 7.1 below) in samples taken from the tendons which were de-tensioned for strand sampling. The results indicate that the corrosion protection medium is not degrading and continues to effectively perform its function. This is borne out by the results of the visual inspections and testing of anchorage components and withdrawn strands (Table 7.2 below), which have shown no evidence of significant corrosion.

Characteristic	Acceptance	Method
Water	5%	Karl Fischer (measured per weight %age)
Chloride	10 ppm	ASTM D512
Nitrates	10 ppm	ASTM D4327
Sulphides	10 ppm	ALPHA 4500
Sulphates	10 ppm	ASTM 4327
Reserve Alkalinity	ASME XI table IWL2525-1	ASTM D947

Table 7.1 Grease Testing

Characteristic	Acceptance
Breaking Load	BS 5896
0.1% Proof Load	BS 5896
Modulus of Elasticity	BS 5896
Elongation at SCUTS (Specified Characteristic Ultimate Tensile Strength)	<4% on 600mm gauge length (as per design specification)
Surface defect on strand	<1.5% of strand wire diameter (as per design specification)

Table 7.2 Mechanical Testing of Strand

(iv) Liner

7.256. A segment of the liner surface from working floor level of dome apex is subject to general visual examination during each refuelling outage. This scope includes both liner plate surface and liner plate welds. The areas of liner surface are inspected either directly or via boom camera (generally for penetration sleeves) or by binocular (generally for dome welds).

7.257. ASME XI IWE requires that the surface examination coverage exceeds 80% accessibility of the nominated containment boundary which provides some practical allowance for areas of liner inaccessibility.

7.258. The general condition of the liner and its paint coating has been found satisfactory. Visual inspections to date have confirmed the absence of significant degradation or penetration of the protective paint coating. Minor indications on the liner plate and coating have been noted but generally consist of small paint chips and scratches associated with the storage or movement of equipment during outage activities. The indications are not considered to be significant and do not threaten the integrity and overall leak tightness of the liner.

7.259. In addition, regular inspection of structures internal to the containment takes place, specifically to detect and assess any signs of damage due to exposure to boric acid.

7.260. A regular routine has been established, whereby these inspections are carried out at each outage being performed to an approved station procedure. This requires a visual inspection to be carried out, with particular attention being paid to any evidence of:

- Leakage of borated liquids from plant.
- Past exposure to borated liquids/vapours.
- Damage to structural elements and/or protective coatings due to current or past exposure to borated liquids/vapours.

7.261. The reporting requirements include measurement and recording of observations, together with the nature and location of any indications of damage or degradation.

7.262. It is noted that the operational regime in relation to boric acid exposure to date has not raised any major threats to the integrity of the civil structures. The exception is the exposure of the primary containment liner and internal structures to boric acid. This threat has been recognised from operating performance and is managed by a Boric Acid Surveillance Programme.

7.263. A comprehensive boric acid examination programme is in place for all components which could be affected, (noting that similar issues have arisen for other PWRs) and judged to be best practice. This includes the civil structures, with regular inspections controlled by an approved station procedure. This is considered to demonstrate good practice, and is an identified strength in the overall ageing management process.

(v) Interactions of liner and concrete containment (Equipment Hatch and penetrations)

7.264. The leak tightness interaction between the liner and concrete including penetrations within the scope of the Containment Leak Rate Testing Surveillance Programme are subject to Type B or C local leak rate tests. Type B tests are on penetration seals and closures and Type C tests are performed on CIVs.

7.265. The local leak rate tests give a final as-left summation of maximum pathway leakage rate for comparison against Technical Specification limits. Where individual tests fail to meet require levels, remedial maintenance is carried out before retesting to confirm integrity.

7.266. An ILRT is performed at regular intervals to determine the total leakage of the PCCV and thus confirm the structure's ability to contain an overpressure which might arise during certain fault conditions.

7.267. The most recent repeat ILRT was undertaken in 2013. The results were well within the acceptance limits, and demonstrate both a very acceptable margin against the allowable limit, and continuing low levels of leakage for the containment. On the basis of the results obtained, the test can remain on the extended test interval of nominally 180 months.

7.268. The Equipment Hatch surfaces including pressure boundary welds are subject to general visual examination. Pressure retaining bolts are not generally disassembled except where door closure bolts are already unfastened and ASME XI VT-3 examination can be undertaken to assess structural material condition. To date, the surfaces, structural and attachment welds and bolts have been found in satisfactory condition. A number of small localised paint chips have been observed on components which come into contact during normal use but are not indicative of an underlying structural or material ageing issue.

7.269. Similarly, the general condition of the electrical and mechanical penetrations (including the sleeve surfaces and welds) are subject to examination. There have been no indications of significant degradation or damage on the penetrations. There have been isolated instances of damage to the paint coating on surfaces including observations of blistering, cracking and flaking paint. Most of the paint remains adherent but could detach and expose the underlying steel. These defects do not pose an immediate threat to the containment capability but are remedied to provide long term protection.

(vi) Seals (moisture barrier at barrel to basemat junction)

7.270. A general visual examination of the moisture barrier at the liner wall to floor slab junction is routinely performed for the Reactor Building circumference during refuelling outage surveillance. The complete moisture barrier is usually visible either for close examination, or at a distance for portions behind plant and equipment. The findings to date confirm the satisfactory condition with insignificant cracking noted in the painted coating surface.

7.1.3.2 Hinkley Point C (NNB)

7.271. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

7.272. These aspects will be developed later for EPR^{TM} UK HPC. Nevertheless, at the date of production of this report, some details can be provided on the EPR^{TM} UK HPC Inner Containment monitoring system (EAU) as well as on the containment leak and resistance tests.

Reactor Building Inner Containment monitoring system (EAU)

7.273. The EAU monitoring system has the following roles:

- At the end of the construction, to provide measurements to confirm that the Reactor Building Inner Containment behaviour complies with the design specifications and requirements.
- Checking of the containment behaviour during pre-stressing,
- Checking of the mechanical behaviour of the structure, in accordance with elastic design assumptions, in Initial Structural Integrity Tests and periodic tests (every 10 years),
- Monitoring durability and structural ageing effects, to check the ability of the containment to fulfil its safety function as the third safety barrier, such as: shrinkage, creep of concrete, relaxation of tendons during the Nuclear Power Plant lifetime,

- EAU system has also a very specific operational requirement regarding the follow-up of the pre-stressing system:
    - Based on the data acquisition relative to concrete strains (from acoustic strain gauges), EAU system enables to detect potential tendon ruptures that could be structurally significant
    - Data acquisition of both strains from acoustic strain gauges and forces from dynamometers enables to detect any atypical delayed pre-stressing losses

7.274. The EAU system comprises of:

- 455 acoustic strain gauges embedded in the concrete
- 225 thermal probes
- 32 vibrating wire sensors within 4 dynamometric wedges
- pendulums on the cylindrical part of the inner containment
- 3 invar wires on the cylindrical part of the inner containment
- 20 topographical survey markers on the upper slab of the pre-stressing gallery
- 24 topographical markers on the raft inside the surrounding buildings
- 17 levelling pots embedded in the common raft and 1 external reference pot
- 11 inclinometers
- 12 displacement sensors in the vicinity of the equipment hatch (only for pressure tests)
- 4 pressure sensors
- 4 ambient temperature probes

7.275. It shall be noted that all tendons are cement grouted except for four vertical tendons which are greased. The tension in the greased tendons is monitored directly using dynamometers located at the upper end of the tendons. Tendon pre-stressing losses can only be measured with sufficient accuracy if the friction between the tendon and duct is very low. This is the case for vertical tendons. In case of horizontal and gamma tendons the friction is too high for accurate tendon dynamometer monitoring. Length variations are measured by eight acoustic strain gauges similar to those embedded within the concrete, distributed along the circumference of the dynamometer cylinder. These four greased tendons represent a significant sample of the containment wall in which to estimate the loss of pre-stress by relaxation.

EPR^{TM} UK HPC containment leak and resistance tests

7.276. The pre-stressed and reinforced concrete cylindrical wall and dome of the internal containment withstand internal pressure, while leak-tightness is ensured by the metal liner.

7.277. At containment penetrations, structural strength and leak-tightness are ensured by various components such as connecting parts between sleeves and penetrations, the equipment hatch and its seals, the personal airlocks, isolation valves for fluid penetrations, electrical penetrations and blind flanges.

7.278. Before unit starts-up, the containment undergoes a test known as the “acceptance test” or “preoperational test”, comprising:

- A “type A” test consisting of an overall leak-tightness test of the inner containment,
- A mechanical resistance test,
- An evaluation of the external containment leak-tightness, to check that it complies with the Annulus Ventilation System design. The Annulus Ventilation System is a system which collects and filters gas leakages in the annulus space, between the inner and outer walls.

7.279. During plant working life, leak-tightness tests of the inner containment wall are periodically repeated.

7.1.4 Preventive and remedial actions for concrete structures

7.1.4.1 Operating reactors (EDF-NG)

7.280. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

7.281. The primary and secondary containment structures are large passive structures and by a combination of conservative design philosophy, robust construction and proof-testing they are generally resilient to emergent defects, which are primarily slow in their manifestation and detectable.

Preventive actions

7.282. Preventive measures are primarily addressed by the regular surveillance activities and monitoring to verify that the containment structures behave in an elastic and predictable manner under normal operating conditions.

7.283. EDF-NG’s physical surveillance activities confirm the material condition and preventative action is undertaken as appropriate, examples of which are as follows:

(i) Concrete surfaces

7.284. Fine cracks (approximately 0.15mm) in the vicinity of the steam and feed penetrations have been fitted with demountable mechanical strain gauge survey points to allow longer term monitoring of the crack behaviour.

(ii) Pre-stressing system

7.285. Any pre-stressing strand removed for visual and mechanical investigation is replaced with new strands which are tensioned to design level (or appropriate) in accordance with ASME XI.

7.286. New strands are coating in protective grease to provide a similar level of corrosion protecting to the remaining in-situ tendon strands.

7.287. Inspected tendon anchorages are routinely coated with fresh grease as required.

(iii) Liner

7.288. As part of the preventative AMP approach, the liner leak tightness is subject to periodic ILRT every ten years. In addition, LLRTs are undertaken at each refuelling cycle to assess local leak tightness around containment isolation values, penetrations and seals.

7.289. Preventive measures also include repainting any defective surfaces. This is routinely undertaken during refuelling outage inspections to address any minor flaking of the main liner, around steel penetrations or access hatches (where physical damage may have also occurred) and reinstating the surface coating to the drain and instrument tunnel sumps.

(iv) Interactions of liner and concrete containment (Equipment Hatch and penetrations)

7.290. Areas of the Equipment Access Hatch or penetration surfaces where minor surface corrosion or paint loss has occurred are subject to recoating, typically at the next available opportunity to prevent long term degradation.

Remedial actions

7.291. Remedial measures tend to fall within two categories: (i) defects discovered during surveillance; and (ii) safety case improvements. The main remedial works to the containment structures are typically associated with repairs to building gap seals which have deteriorated or been subject to damage.

(i) Liner

7.292. ASME XI requires the following defects to be rectified or evaluated prior to return to service:

- Pressure-retaining component corrosion or erosion that exceeds 10% of the nominal wall thickness.
- Loose, missing, cracked or fractured parts including bolts and fasteners.
- Structural distortion or displacement of parts to the extent that component function may be impaired.

7.293. Localised sections of the drain sump liner thickness have been reduced due to corrosion of the mild steel covering plate. The floor drain sump liner integrity presents a vulnerable area for further corrosion and thickness loss. The mild steel liner plate that covers the floor drain sumps is subject to visual examination and ultrasonic thickness checks of any defects indicative of significant loss of section. To date, the majority of defects have led to limited loss of section thickness and remedial works comprise removal of surface corrosion and application of a new paint protection layer. A similar strategy has been adopted for the instrument tunnel sumps.

7.294. On occasion, where liner thickness has significantly reduced locally, a grid of ultrasonic thickness measurements is taken. Degradation occurs due to breakdown of the paint coating either by physical impact of items in the sump or through exposure to various substances that are present in the sump and residual sediment, including boric acid effects.

7.295. A local leak rate test can be conducted on a through-thickness defect using a partial vacuum technique to avoid pressure loading the underside of the sump floor and causing wider damage. This technique also has the advantage of drawing any water under the floor plate towards the leak site for removal. The leakage rate is determined and compared against ASME XI allowable values.

7.296. New plate is welded onto the sump floor (justified via the safety case modifications process) to reinstate a leak-resistant boundary with preference towards an over-plate solution of the whole base area rather than local patch repairs.

7.297. The AMP would benefit from the sharing of industry experience for the best practice management for these vulnerable areas with particular information on available protective coating systems that could be applied (see paragraph 7.306).

(ii) Seals

7.298. Moisture barriers with wear, damage, erosion, tear, surface cracks or other defects that permit intrusion of moisture against inaccessible areas of the pressure retaining surfaces of the liner containment are repaired at each surveillance campaign prior to return to service from the refuelling outage.

7.299. Inspection of the secondary containment enclosure building gap seals is also routinely undertaken and findings to date confirm the overall integrity, remedial works have included:

- In a number of locations, patch repairs have been made following previous damage to the seal material by protruding bolt shafts.
- Protective rubber shrouds have been fitted to the bolt shafts to prevent damage at floor slab and lower floor levels, but not at higher levels.
- Only minor isolated defects have been observed.
- Where other minor damage to the seals has been noted, it is repaired to prevent the defects becoming more serious and affecting the ability of the seal to perform its function.
- Damage can also occur where inflated seals are rubbing against rough or unfinished ends of adjacent pipe support steelwork or Unistrut, and this can be avoided by wrapping to protect the seal.

7.300. The building seals around access doors and penetrations are inspected on a rolling programme under station civil maintenance. Inspection results and any defects observed are recorded through the Work Management System. Any necessary maintenance work is carried out after Station approval. Thus, the door seals are maintained in a satisfactory condition.

7.301. Similarly, the joints and associated gap seals in the principal fire barrier walls are inspected and any remedial actions undertaken at the time. The inspections are undertaken by station SQEP personnel and recorded in the Work Order Card process on AMS. To date, the inspections have generally concluded the condition of the gap seals to be acceptable.

7.1.4.2 Hinkley Point C (NNB)

7.302. NNB advises that these aspects will be developed later for EPR™ UK HPC.

7.2 Licensees' experience of the application of AMPs for concrete structures

7.2.1 Operating reactors (EDF-NG)

7.303. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

7.304. The comprehensive surveillance measures in place are intended to identify any potentially adverse ageing and degradation processes in a timely manner for correction before they present a threat to the long term integrity of the PWR primary and secondary containment structures.

7.305. The EDF-NG AMP approach relies upon a series of integrated processes that primarily rely upon the robust inspection and surveillance programme. This approach confirms the material condition and integrity of the PCCV and secondary containment to continue their safety function. Adverse findings that are out with the predicted expectations or visual evidence that is obtained for deteriorating material condition are either addressed by routine maintenance or specific improvement measures.

7.306. To date the ageing phenomena and long term structural behaviour of the concrete containment structures has aligned with the design expectations e.g. pre-stressing loads, concrete condition, structural performance during leak rate test results, grease protection, surface paint coatings & seals.

7.307. In general the features recorded to date from the concrete containment surveillances do not represent significant degradation of the condition of the structure and do not raise any concerns for its continuing integrity and fitness for service.

7.308. The type and number of indications and defects observed are also noted to be similar to those observed during previous inspections. This indicates that no significant step changes or increasing degradation in the condition of the concrete structures is taking place.

7.309. The general good condition of the structures and the absence of significant degradation have allowed some changes to surveillance activities that underwrite the AMP to be made including:

- Revision of the six-monthly inspection regime for the secondary enclosure building to a single inspection per refuelling cycle.
- Continued validity of an extended ILRT interval of notionally 180 months.

7.310. The overall AMP surveillance has been enhanced to address external OPEX and findings from the internal inspections as follows:

- Boric acid surveys are undertaken of structures located internal to the containment to assess for degradation effects.
- Greater focus on floor drain sump liner integrity.
- Implementation of demountable mechanical strain gauge measuring points to monitor concrete crack behaviour in the vicinity of the high temperature steam / feed penetrations.
- Recommendations by the Appointed Examiner to improve the reliability and performance coverage of the embedded concrete instrumentation (VWSGs and RFSGs) to assist lifetime assessment.

7.311. The main nuclear safety-related structure comprises the primary concrete containment (PCCV) forming the main Reactor Building and this is sheltered from the external environment by the Secondary Containment Enclosure Building and Auxiliary Building. Therefore, it is not subjected to exposure to the weather and in terms of the AMP it is considered unlikely that any significant change in the current structural condition will occur.

Recommendations by EDF-NG

7.312. It is judged that the current strategy and measures in place provide an adequate AMP for the PWR concrete containment structures. EDF-NG continues to seek an improved understanding of the ageing mechanisms and this review has identified some common industry areas where wider information-sharing could provide benefit to the longer term ageing management strategy for the PWR concrete containment structures. It is therefore proposed that the following topics are raised for discussion during the 2018 topical peer review workshops:

- Utility best practice-sharing on the maintenance, refurbishment and replacement of concrete strain measuring devices to support long life surveillance, taking cognisance of improvements in technology and remote monitoring.

- Utility guidance on improving the long term integrity of the sump drains liners with information regarding best practice ageing management to maintain liner thickness including applied coating protection systems.
- Utility experience on the long term exposure of steel surfaces and components to a boric acid environment.
- Utility experience of ground water ingress affecting the bottom liner with advice on enhanced groundwater sampling and preventative measures.

7.2.2 Hinkley Point C (NNB)

7.313. The text in this section of the report has been prepared by NNB, with only minor changes as described in paragraph 1.9 to 1.10.

7.314. External OPEX has been taken into account to inform NNB's management of ageing for the EPR^{TM} UK HPC. For instance:

- Concrete exposure to sulphate attack: consideration of degradation of civil structures located in the Severn area has informed the protection of the EPR^{TM} UK HPC structures
- Durability of geo membrane: review of the performance of German and Swiss membranes installed at the outside of rail tunnels has supported the durability assessment of the EPR^{TM} UK HPC membrane.

7.3 Regulator's assessment and conclusions on ageing management of concrete structures

7.3.1 Criteria and standards for assessing ageing management of concrete structures

7.315. The relevant criteria and standards adopted within this assessment are principally the Safety Assessment Principles (SAPs) and ONR Technical Assessment Guides (TAGs), together with relevant national and international standards and relevant good practice informed from existing practices adopted on UK nuclear licensed sites. In addition to the general criteria and standards described in Section 2 of this report, ONR has utilised the following in the assessment of ageing management of concrete structures:

- Relevant ONR Safety Assessment Principles on civil engineering (Ref 3)
- ONR Technical Assessment Guides:
    - Civil Engineering (Ref 52)
    - Civil Engineering containments for reactor plant (Ref 53)
- International Atomic Energy Agency:
    - Safety Report Series 82 - Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (Ref 54)
    - Nuclear Energy Series NP-T-3.5 - Ageing Management of Concrete Structures in Nuclear Power Plants (Ref 55)
- American Society of Mechanical Engineers - Boiler Pressure Vessel Code:

    - Section III - Rules for Construction of Nuclear Facility Components - Division 2 - Code for Concrete Reactor Vessels and Containments (Ref 56)
    - Section XI - Rules for In-service Inspection of Nuclear Power Plant Components (Ref 57)
- French Society for Design, Construction, and In Service Inspection Rules for Nuclear Island Components (AFCEN):
    - ETC-C (2010) - EPR Technical Code for Civil Works (Ref 58)
- Electric Power Research Institute (EPRI):
    - Program on Technology Innovation: Concrete Civil Infrastructure in United States Commercial Nuclear Power Plants (Ref 59)

7.3.2 ONR's assessment of EDF-NG's ageing management of concrete structures

7.316. EDF-NG's methods for selecting components within the scope of its AMP are defined in the safety case, which identifies structures, systems and components (SSC) important to safety and the associated claims on the integrity of the concrete structures. Based on the safety claims, the safety case identifies requirements for examination, inspection, maintenance and testing (EIMT). ONR regularly inspects the adequacy of EDF-NG's arrangements for EIMT of these concrete structures and considers that the methods used to select SSC for inclusion in its AMP are adequate.

7.317. ONR has also considered relevant IAEA guidance with respect to ageing management programmes (Ref 54 and Ref 55) and has identified one omission from the licensee's AMP. This omission relates to the embedded strain gauges used to monitor the containment. Although the gauges are not claimed in the safety case, they provide information of relevance to ageing management of the concrete structures and ONR consider this omission an area for improvement as described in Section 7.3.4.

7.318. The main procedures used by EDF-NG to identify ageing mechanisms for the concrete structures are a set of station-specific "Rules for In-Service Inspection of the Civil Engineering Works". The licensee's "Rules" are based on the requirements identified in ASME XI (Ref 57), which ONR considers an appropriate technical standard for the inspection of concrete containment structures. The inspection requirements are compatible with the ASME-based design standards for the structures. In addition to visual inspections, the "Rules" also cover the periodic leakage rate testing programme. ONR has seen evidence from its inspections that the "Rules" are regularly reviewed in response to both operating experience and observed ageing effects.

7.319. An Appointed Examiner (APEX), who is independent of station operations and the safety case justification, manages the civil engineering surveillance programme and reports on their findings. ONR considers that the use of an independent suitably qualified and experienced APEX in this role is in accordance with relevant good practice. As part of its permissioning regime under the licence conditions, ONR regularly assesses the adequacy of the APEX's examination reports for the concrete structures. Based on these assessments, ONR is satisfied that the APEX's surveillance programme identifies ageing mechanisms in the concrete structures and that EdF-NG's process for appointing the APEX results in suitably qualified and experienced persons being appointed.

7.320. EDF-NG invokes its Safety Case Anomalies Process (SCAP) where an examination, inspection, maintenance or testing activity reveals a plant condition that challenges the ageing and degradation assumptions within its safety case. Where these challenges are significant, the licensee may need to make modifications to the safety case, using the design change process to justify continued operations. For the concrete structures, ONR has experience of assessing the safety justifications produced by EDF-NG when changes to expected conditions have been identified (for example in relation to the level of corrosion in tendons). ONR's experience is that the SCAP and design change processes are adequate for capturing ageing-related changes in the concrete structures that are outside the safety case parameters.

7.321. EDF-NG has described the outputs from its ageing assessments, including the identification of ageing mechanisms, their significance and the applicable acceptance criteria. The conclusions of ONR's assessment of the evidence relating to EDF-NG's ageing assessment of the concrete structures are given below for each case identified for inclusion in the NARs.

7.322. The focus of EDF-NG's approach to identification of concrete degradation is visual inspection. ONR has seen only limited EDF-NG guidance to help its inspectors identify the ageing mechanisms likely to lead to cracking. No specific acceptance criteria for concrete cracking have been identified, other than a general requirement for the primary containment structure to pass periodic Integrated Leak Rate Tests (ILRT). ONR's experience is that the APEX initially assesses concrete defects for their structural significance and when required, consults with PCCV designers who would recommend any necessary remedial works. Given the general lack of significant concrete defects, ONR considers the approach taken to be acceptable. Although EDF-NG's assessment does not specifically mention shrinkage and creep effects, ONR's experience is that the APEX periodically assesses trends in concrete strain, as measured by embedded strain gauges, to determine whether these ageing phenomena are occurring in line with design (and safety case) assumptions. ONR's assessments have concluded that these effects are being adequately monitored.

7.323. EDF-NG identifies reinforcement corrosion as an ageing mechanism but considers the risk to be low, due to the protection provided by the surrounding concrete and the limited exposure of the primary and secondary containment to chlorination and carbonation effects. ONR's experience from inspections is that the containment structures do not show signs of reinforcement corrosion. Acceptance criteria for reinforcement corrosion are not explicitly stated, though the licensee considers that the visual inspection programme will identify wider reinforcement corrosion of the primary containment on both concrete faces. The significance of any observed corrosion would be assessed in the first instance by the APEX. Given the lack of evidence of reinforcement corrosion ONR judges that EDF-NG's approach is adequate.

7.324. EDF-NG has identified, in general terms, the important ageing mechanisms affecting the pre-stressing systems. Although little detail is given regarding specific ageing mechanisms, ONR considers that the APEX community within EDF-NG has significant knowledge in this area, which has been informed by operating experience. Detailed acceptance criteria based on ASME requirements are used for the tendon load tests and material tests on strand and its protective grease. For the anchorages, acceptance is based on the judgement of the APEX that there has been no damage or degradation sufficient to warrant further evaluation. ONR considers the pre-stressing acceptance criteria to be well-defined and adequate.

7.325. EDF-NG has identified appropriate ageing mechanisms likely to affect the liner and its protective coating. These mechanisms also include consideration of exposure to boric acid resulting from plant leakages. The liner is attached to the concrete shell by an arrangement of structural anchors. The anchors cannot be physically inspected and reliance is placed on design margins and the presence of any secondary indications of distress from the detailed surface examinations of adjacent concrete and liner. ONR has not observed any secondary signs of distress in the anchors as part of its inspections. The acceptance criteria for the liner and liner anchor defects are based on the judgement of the APEX; this is in line with ASME guidance and ONR considers the approach to be acceptable.

7.326. EDF-NG has not presented any evidence that it has identified the relevant ageing mechanisms for the four types of seal that are subject to ageing management. EDF-NG has also not presented any acceptance criteria for seal defects. Reliance is instead placed on its inspections to identify defects and specify any necessary maintenance work, though it is not clear to ONR what guidance the inspectors use to assess the significance of any defects found. ONR considers this to be an area for improvement, as described in Section 7.3.4.

7.327. A technical surveillance programme is developed for each refuelling outage, which takes place every 18 months. The programme allows the scope of work to be adjusted to reflect operating experience. The APEX reports the results of the surveillances and inspections in line with refuelling outage processes. The APEX's report supports the licensee's return to service considerations. ONR has assessed the APEX's "In Service Inspection Summary Report" for each outage as part of its permissioning process under the licence conditions.

7.328. Detailed surveillance of the pre-stressing system is undertaken on-load every five years and reported in the licensee's inspection report for the next refuelling outage. Integrated leak rate tests (ILRT) are undertaken every 15 years. ONR considers that the surveillance intervals reflect satisfactory previous test performance and operating experience and are adequate.

7.329. ONR's assessment of EDF-NG's surveillances has concluded that:

- The codes and standards used during the surveillances (predominantly ASME XI) are adequate and appropriate for this work.
- The examination and test procedures are appropriate to the safety classification of the structures.
- The surveillances have adequately demonstrated the absence of significant defects or, where defects have been found, adequate safety arguments have been made for the next period of operation.
- The measured pre-stressing tendon anchorage loads were greater than the minimum required values for each group of tendons. ONR is content to accept the APEX's conclusion that the tendon loads are predicted to stay above the minimum requirement until the next tendon load checks.
- ONR has accepted the judgements of the APEX that the concrete structures are in a satisfactory condition for return to service.

7.330. ILRT are conducted at defined intervals throughout the operational life of the vessel. The test verifies that the leakage from the primary containment under certain postulated post-accident conditions does not exceed the maximum leakage allowed by the safety case. The test aligns with the expectations in ONR's SAPs and accords with relevant good practice. From its review of the APEX's assessment and its observations of the last test, ONR considers that the ILRT is providing useful information regarding ageing management and that results to-date have confirmed the continuing adequacy of the primary containment.

7.331. In addition to the ILRTs, the effects of ageing are assessed using local leak rate tests. ONR considers that the results have shown a declining trend in overall leakage rate and show an adequate margin to safety limits, indicating that ageing effects have not led to a reduction in the leak tightness of the primary containment.

7.332. ONR's experience from the assessment of EDF-NG's safety submissions is that effective use is made of trending of parameters that may be expected to vary with time due to ageing effects. An example is the loads in the pre-stressing system. ONR considers that the change is tendon loads has followed the expected decrease with time, whilst still demonstrating adequate margins with respect to the safety case limits.

7.333. ONR's experience from inspections and assessments is that where the licensee identifies defects, they are either routinely addressed during shutdowns (for example repainting to damaged areas of liner coating) or are raised as APEX recommendations to be addressed at future shutdowns.

7.334. In terms of criteria for taking action when defects are identified, these range from the prescriptive requirements of ASME XI for liner defects, to the use of judgement by suitably qualified and experienced persons regarding the monitoring of concrete crack widths and the assessment of damage to seals. In taking remedial action, EDF-NG follows its modifications process (a licence condition requirement), the implementation of which is regularly assessed by ONR and considered acceptable.

7.335. EDF-NG has identified a limited number of vulnerable areas where liner corrosion has occurred, in particular in the floor drain sumps. ONR is satisfied that these areas are being appropriately monitored, and where the liner thickness has reduced below the acceptance criteria, that suitable repairs have been carried out.

7.336. Overall, ONR considers that the actions taken by EDF-NG in response to the identification of defects have been informed by the safety case and operating experience and have been adequate.

7.3.3 ONR's assessment of NNB's ageing management of concrete structures

7.337. The detailed methods used by NNB to select components within the scope of the AMP are still to be developed. Guidance in IAEA NS-G-2.12 (Ref 12) suggests that an AMP appropriate to the design and procurement stages of construction should already be in place.

7.338. The safety case identifies the SSC important to safety and the associated claims on the integrity of the concrete containment structures. Based on the safety claims, ONR's expectation is that the safety case will identify an AMP for the concrete structures, including identification of SSC to be included within it. The safety case for concrete structures is currently at the Pre-Construction stage and is being assessed by ONR. As part of its assessment, ONR is considering the extent to which ageing phenomena have been addressed in the design of the concrete structures.

7.339. A licence condition requires the licensee to make and implement adequate arrangements for the regular and systematic examination, inspection, maintenance and testing (EIMT) of all plant which may affect safety. Detailed EIMT arrangements are still being developed and have not yet been seen by ONR.

7.340. A preliminary list of inspection activities has been identified by NNB, which ONR considers covers the majority of items that it would expect to be included in a future AMP. Whilst acknowledging that detailed inspection arrangements are still to be developed by NNB, ONR notes that the preliminary list does not identify any detailed inspections of the liner. ONR considers that addressing the areas for improvement described in Section 7.3.4 should assist NNB with the development of appropriate liner inspection plans.

7.341. NNB has not described any specific process or procedure for the identification of ageing mechanisms for concrete structures. Reliance is instead placed on the design process to take account of ageing phenomena. The designs are assessed against NNB's Nuclear Safety Design Assessment Principles (NSDAPs), which require the design to take account of ageing in setting design margins, allow for ageing in the design and make provision for monitoring, testing, sampling and inspection to assess ageing mechanisms. The design code used for HPC is based on ETC-C (Ref 58). The code requirements are based on an assumed design life of 80 years for the concrete structures.

7.342. ONR considers that the use of appropriate design codes results in the majority of ageing phenomena being accounted for at the design stage. It would nevertheless be beneficial for NNB to have a procedure that would assist the designer in formalising their assumptions regarding ageing into an Ageing Management Programme that could then be further developed after the construction phase and used to identify detailed EIMT requirements. ONR considers the lack of a procedure for ageing management to be an area for improvement, as described in Section 7.3.4.

7.343. NNB has described the outputs from its ageing assessments, at the design and procurement stages, including identification of ageing mechanisms, their significance and the applicable acceptance criteria. The conclusions of ONR's assessment of the evidence relating to NNB's ageing assessment of the concrete structures are given below for each NAR example.

7.344. NNB has assumed a working life of 100 years for the concrete design, which provides an acceptable margin to the minimum working life of 80 years used in the design code. An appropriate allowance has been made in the design for the effects of ageing mechanisms associated with exposure to the environment and to aggressive soil conditions. ONR agrees that a suitable test programme has been undertaken in order to confirm:

- that proposed concrete mixes do not contain deleterious levels of aggressive elements such as chlorides and sulphates that could lead to accelerated deterioration of the structures; and
- that design assumptions for ageing related phenomena such as creep are valid.

Protection of the reinforcement against ageing effects relies on the suitability of the concrete mix, the adequacy of the cover to the reinforcement and the quality of the construction. In regard to ageing management, ONR considers that the concrete mixes and the specified covers meet the requirements of the appropriate standards and are adequate.

7.345. The ageing-related design and procurement requirements for the pre-stressing strands, such as the limits on relaxation and acceptable corrosion resistance, are included in a specification and are appropriate. The pre-stressing anchorage components that are not embedded in the concrete will be protected with an appropriate coating. The pre-stressing strands are reliant in service on being fully surrounded by a cementitious grout. ONR considers that the adequacy of the grout materials and workmanship specification are important to protect the strand from ageing effects and has concluded that:

- NNB’s grout specification sets the criteria to limit aggressive elements such as chlorides and sulphur to acceptably low levels
- The specification requires grouting tests in advance of nuclear safety-related construction, including full-scale mock-up tests on each tendon type to confirm that the ducts can be filled with grout. These tests are in accordance with applicable standards and have clear acceptance criteria.

7.346. Material specifications for the liner and other steel sections are adequate. NNB has not provided any information for the protective coating to the liner in its NAR contribution. The specification in ETC-C says simply that it is to be corrosion-resistant and subject to approval before fabrication. ONR considers this to be an area for improvement, as described in Section 7.3.4.

7.347. Limited information is given regarding the ageing assessment and acceptance criteria for the seals at building penetrations, other than a statement that they will need to retain their mechanical properties for a minimum of ten years and not be subject to dimensional variation due to ageing. ONR considers this to be an area for improvement, as described in Section 7.3.4.

7.348. Although the proposed testing, sampling and inspection activities post-construction have still to be finalised, there are still activities being undertaken before and during construction that are relevant to ageing management. ONR has carried out a series of inspections and assessments in relation to NNB’s development of suitable concrete mixes that address relevant ageing and degradation mechanisms. The findings of ONR’s inspections and assessments are that NNB has utilised relevant good practice to develop and implement a thorough range of pre-construction tests and trials and has adequate arrangements in place to design, specify and produce concrete mixes for nuclear safety related construction.

7.349. NNB is designing a containment monitoring system to provide information of relevance to ageing during commissioning, structural integrity testing and operation. The system will also be capable of detecting a potential tendon rupture due to ageing or other phenomena. ONR considers that the proposed monitoring system is in accordance with relevant good practice.

7.350. Although nearly all the tendons are protected from corrosion by grout, four vertical tendons are not grouted and will be used for monitoring purposes. The monitoring will measure the loss of force in the tendons due to ageing effects so that it can be compared with design assumptions. ONR considers that the proposed monitoring system is in accordance with relevant good practice.

7.351. The licensee has given outline details of proposed leak tests on the primary containment and its penetrations. ONR considers that the basics of a leak-testing programme have been identified that is sufficient for this stage of the design process.

7.3.4 Overall conclusions on ageing management of concrete structures

7.352. Based on its assessment, ONR considers that both EDF-NG’s and NNB’s SSC-specific AMPs for concrete structures are adequate. ONR accepts that the AMPs reflect the different lifetime stages of the relevant stations, with NNB’s AMP being less developed than that of EDF-NG, but appropriate for the design and procurement stages of construction

7.353. EDF-NG has raised four recommendations identifying where it considers that wider information-sharing could provide benefit to its longer term ageing management strategy and which should be discussed during the Topical Peer

Review during 2018. ONR supports EDF-NG's recommendations and considers that any learning from this review should be included in revised EDF-NG company guidance as identified in the areas for improvement. The areas for wider information sharing identified by EDF-NG are:

- maintenance, refurbishment and replacement of concrete strain measuring devices;
- long term integrity of the sump drains liners;
- long term exposure of steel surfaces and components to a boric acid environment; and
- ground water ingress affecting the bottom liner with advice on enhanced groundwater sampling and preventative measures.

7.354. ONR's assessment has identified a number of strengths in EDF-NG's AMP, in particular, the importance it gives to the role of the independent APEX and the adoption of a surveillance programme that is in accordance with relevant good practice and is regularly reviewed and adjusted to reflect operating experience.

7.355. ONR considers the main strengths of NNB's SSC specific ageing management programme for concrete structures are:

- The surveillance programme for the concrete mix design and trials, which should minimise the risk of deleterious ageing effects in the finished structures.
- Its extensive arrangement of monitoring devices to be installed on the primary containment so that key parameters that may be affected by ageing can be measured and compared with design assumptions.

7.356. NNB has not identified any areas for improvement in its SSC-specific AMP.

7.357. In spite of having adequate AMPs, this assessment has identified a number of areas for both licensees where improvements to ageing management would be beneficial. For the concrete structures, these are as follows:

- EDF-NG should provide guidance on the ageing management of PCCV strain gauges by the end of 2018 (paragraph 7.317).
- EDF-NG should review its company guidance relating to ageing mechanisms and acceptance criteria for concrete structure SSCs by the end of 2018 to ensure that it is comprehensive and based on relevant good practice (paragraph 7.326).
- NNB should formalise its ageing management arrangements for concrete structures, for all stages of construction, by the end of 2018. The arrangements should describe the methods and criteria used for selecting the SSCs to be included in the programme and the processes/procedures for the identification of relevant ageing mechanisms (paragraph 7.342).
- NNB should review ageing mechanisms and acceptance criteria for the liner coating and penetration seals, to ensure that these are comprehensive and provide appropriate guidance to enable the development of the material specifications and EIMT activities. The review should be completed by the end of 2018 (paras 7.346 and 7.347).

7.358. To ensure that these improvements are implemented in a timely manner they have been brought together along with those from other chapters in Chapter 9. Chapter 9 identifies a single area for improvement (AFI) for each licensee to undertake a programme of improvement and the individual elements necessary to complete that programme are clearly identified, along with dates for their completion.

7.359. The overall conclusion from the assessment of concrete structures is that both licensees have adequate AMPs, given the specific stages of the lifetimes of their plants, but that both licensees need to make a limited number of secondary beneficial improvements.

8 Pre-stressed concrete pressure vessels (AGR)

8.1 Description of ageing management programmes for PCPVs

8.1.1 Scope of ageing management for PCPVs

8.1. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

Introduction and description of structures

8.2. The continuous safety of the Pre-stressed Concrete Pressure Vessels (PCPVs) is assured by compliance against the duties and arrangements set out in the Nuclear Site Licence Conditions.

8.3. The EDF-NG fleet of seven AGR power stations (see Section 1, Table 1) each contains two PCPVs (see Figure 8.1). These structures are subject to routine monitoring and surveillance as a key part of the overall ageing management programme (AMP).

8.4.

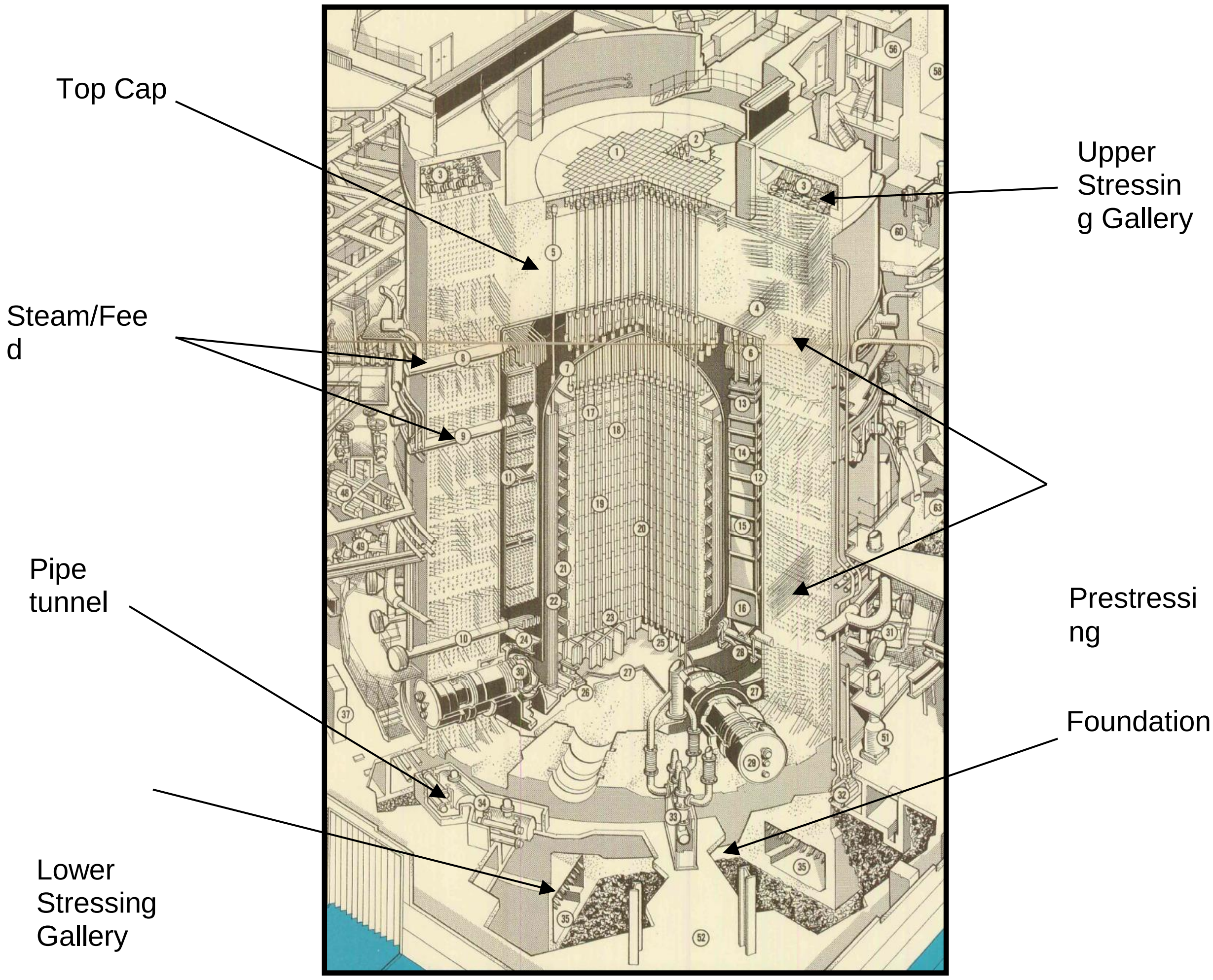


Figure 8.1 Cut-away cross section of typical AGR PCPV

8.5. The PCPVs are required to remain safe to operate for their lifetime requirements and timely surveillance helps to identify any material condition shortfalls that could preclude this objective from being achieved.The AMP ensures the safety functionality and duty of the PCPVs are not affected by ageing mechanisms during the lifecycle of the station.All of the elements that make up an AMP for the PCPV structures are either embedded in the station safety cases or are part of the company procedures and processes. The general ageing management arrangements are discussed in Section 2, and will not be repeated here. Section 8.1.1 onwards will focus on the specific arrangements for the PCPV structures.

8.6. The Maintain Design Integrity (MDI) process controls the modification, maintenance and documentation of the safety case. It ensures that the safety case claims on the PCPV structures and its associated documentation remain consistent with each other and with the design intent. This includes the specification of a MS, which covers testing and inspection of the structures in order to demonstrate that their condition remains aligned with the safety case.

8.7. The supporting analysis to the safety case assesses the structural ageing and degradation in setting appropriate operating limits and the inspection and testing regimes. For significant lifetime issues this may also include identification of appropriate research and development to underwrite ageing and degradation and other modelling assumptions.

8.8. Within EDF-NG, the Design Authority role is carried out by the Design Authority organisation (under the Head of Design Authority). Several other parts of the EDF-NG organisation may be required to support the Design Authority in the delivery of the its role e.g. Plant Discipline SQEP(s) where specific technical knowledge and authoritative advice is required (and resides out with the Design Authority). This arrangement is embedded within a Company Specification.

8.9. In the case of the PCPV structure, the PCPV Design SQEP role is enacted by civil engineering SQEPs to provide technical and authoritative advice for safety issues affecting the PCPV safety case, integrity, operation, review etc. Proposed changes to the design envelope, operating function or monitoring of the PCPV must be undertaken in accordance with the safety case modifications process.

8.10. Authoritative technical advice on the PCPV is provided by SQEP personnel who can fulfil the PCPV Design SQEP role and subject to independent review and agreement by the nominated Appointed Examiner. Any structural defects, non-compliances and/or ageing issues that may be identified by the Appointed Examiner are subject to review and assessment by the PCPV SQEP in accordance with the company governance arrangements; this ensures that ageing issues that could affect the safety case envelope are adequately addressed.

8.11. PSRs are carried out every ten years in accordance with the licence condition for periodic review (paragraph 2.48). For the PCPV structures there have been two or three such full reviews to date (depending on the station age) with the current PSR reports providing confidence that the PCPV structures continue to be safe as judged against the safety standards and practices in force at the time. The main objectives of the PSR reviews are to:

- Validate that the physical conditions and capabilities of the PCPV remain in accordance with the Safety Case requirements. This includes the structural integrity under both normal operational conditions, and against the identified internal and external hazards.
- Confirm the adequacy of the PCPV for operation at least for the period until completion of the next PSR

- Review relevant developments in research and development activities, and national and international experience.
- Validate that any ageing and degradation mechanisms or other factors (e.g. obsolescence) that could affect future safe operation are reflected within the safety case, or that measures are in place to ensure they are duly considered.

These objectives align with the requirements of AMP.

8.12. Looking ahead to PSR3, the review strategy will focus more upon challenging the generic arrangements on how the SSCs are managed. This will primarily consider the effectiveness of technical governance, plant health committees, peer group improvement plans, equipment reliability reviews (including SSRs), system action plans and the maintenance schedule adequacy. Hence the integrated processes that are embedded across the business will be reviewed in detail to assess their continued contribution to the overall reliability improvement for the SSCs. This approach still aligns with the broader AMP remit whereby many of the key initiatives (e.g. TLMS) and decision influencing (e.g. risk management) become embedded into normal business.

8.13. Due to the passive nature of the PCPV structures (similar to many large civil structures), the AMP surveillance and condition trending activities do not strictly align with the company Equipment Reliability (ER) processes. The ER process is focused predominantly upon plant and system components where active changes in plant configuration & performance can be readily measured and recourse to condition monitoring methods can be used to generate informative metrics. The long term condition and trending of the PCPVs is included in the existing SHIP processes. However, the output offers only a limited overview of the actual structural performance and understanding required to provide relevant information that can inform the forward AMP strategies. Instead, the long term performance of the PCPVs is more efficiently informed by the detailed MS programmes for compliance with the licence condition for EIMT and associated reporting. This focused approach more than counters any shortfalls in trying to fully align a plant orientated ER process to large and passive civil structures.

8.14. EDF-NG implements Technical Governance (TG) and oversight arrangements for the PCPV structures to provide assurance that these key structures are appropriately managed. A top level company strategic policy for the PCPV structures is supported by two mandatory Company Technical Standards:

- Firstly, for the management of concrete pressure vessels – a company standard specifies the requirements for safe management of the PCPVs and sets down the appropriate governance arrangements.
- Secondly, to address water leaks emanating from the embedded Pressure Vessel Cooling Water (PVCW) pipework and CO_2 leaks from the pressure boundary that could affect the PCPV integrity – a separate company standard provides the reporting and assessment routes to be followed in the event of leaks including the maintenance of a records database recording all such leaks (and associated remedial actions) across the AGR fleet.

8.15. Furthermore, an additional company document provides the guidance on compliance with the Pressurised Systems Safety Regulations (PSSR 2000) which includes the PCPV operation within its legal scope.

8.16. The governance of the PCPV design codes is addressed via a company guidance document which sets down guidance for the use of civil engineering design codes, including BS 4975 1990: Pre-stressed Concrete Pressure Vessels for Nuclear Engineering.

8.17. Structural reliability and performance related risks associated with the PCPVs are addressed in the same manner as all other plant risks via the company risk log and risk management process. This ensures that specific risks associated with the PCPVs are identified, understood, valued and appropriate mitigation strategies are developed. The risk profile for the PCPVs is continuously monitored and reported to senior and executive management. The risk structure for PCPVs is primarily represented by the following top level risks (which act as headings to organise the significance and relevance of potential risks):

- Failure of liner and penetrations.
- Failure of tendons/ stressing system.
- Failure of concrete.
- Failure of foundations.
- Failure of PVCW supply.
- Failure of safety relief valves.

8.18. The risk values both in nuclear safety significance and commercial risk are reviewed quarterly throughout the year and the current risk position and mitigation progress is reviewed and reported to senior management. The risk management process aligns with the ageing management principles to ensure visibility of life limiting issues and to support prioritised investment.

8.19. The wider remit of governance across the fleet for critical plant items includes the establishment and ownership of a Through Life Management Strategy (TLMS). A TLMS has been developed for the PCPVs across the fleet and the strategy focuses on the components most vulnerable to age-related degradation. It systematically identifies the activities needed to achieve the lifetime requirements of the structures and associated safety components. The top level aims of the TLMS are to:

- Collect together a number and range of PCPV related initiatives and strategies under a ‘single banner’.
- Develop a standardised approach to safely achieve lifetime requirements.
- Provide senior management and the investment decision makers with visibility and an understanding of the investment landscape.
- Develop a methodology to prioritise investment decisions that support ageing management of the PCPVs.

8.20. The PCPV strategy is divided in three sections:

- Reviewing the current maintenance, inspection and testing regimes.
- Development of fleet wide performance improvements.
- Support resolution / mitigation of risk prioritised plant issues.

8.21. The maintenance, inspection and testing is considered to be discharged as part of ‘normal business’ and provides the foundation for the overall AMP. The remaining two sections are of equal importance to the overall lifecycle requirements of the PCPV.

8.22. Key areas of particular note from the PCPV TLMS relevant to AMP include:

- Forward focus on mitigating the risk to the PCPV lifetime as a result of PVCW and CO_2 leaks.
- Reviewing the effect of partial loss of PVCW affecting PCPV integrity.
- Continuous review of measured in-situ pre-stressing loads (noting the reducing residual pre-stressing loads against minimum levels).
- Ageing of embedded PCPV instrumentation.

8.23. The long term issue of obsolescence of PCPV instrumentation is outside the scope of this report but it is important to note that the output from the instrumentation is important to inform the structural performance of the PCPVs. Hence the instrumentation output provides valuable information that is relevant to the on-going assessment of ageing mechanisms.

8.24. The PCPVs comprise reinforced and pre-stressed concrete vertical cylinders. They constitute massive structures with concrete volumes ranging from 12,000 to 25,000 cubic metres and 250 tons of pre-stressing steel (see Figure 8.1).

8.25. The PCPVs are supported on massive reinforced concrete foundations that transfer the combined weight of the structure, internals and attachments down to an adequate load bearing strata via deep reinforced concrete base slabs bearing directly upon rock, suitable ground strata or via large diameter piled foundations.

8.26. The vessels at Hartlepool/Heysham 1 comprise a podded PCPV design and are essentially vertical cylinders with thickened walls in which are formed eight podded cavities containing the main boilers (see Figure 8.2).

8.27. The main structural elements and components of the PCPVs comprise:

- The PCPV structural concrete: (including embedded cooling pipework).
- The PCPV pre-stressing system: steel tendons and anchorages (including grease protection).
- The PCPV support structure: central foundation disk, support walls, bearings, piled foundations.
- Interfacing civil structures: lower stressing gallery, pipe tunnel, secondary closure restraints (podded PCPV design only).

8.28. The principal safety functions of the PCPV are:

- To ensure a pressure boundary between the primary coolant and the outside environment, by providing support to the PCPV steel liner and penetrations.
- To provide support for the reactor internals, boilers and gas circulators.
- To provide a radiological shield between the reactor internals and the outside environment.

8.29. The AGR PCPV fleet has evolved in design and layout but all essentially comprise pre-stressed and reinforced concrete upright cylindrical structures (see Figures 8.1 and 8.2 for typical examples).

Methods and criteria for selecting components within the scope of the AMP

8.30. The PCPVs are required to maintain their structural performance over required levels (defined in the safety case) with adequate reliability during their design service life. Consequently, inspection and maintenance are an indispensable part of the AMP to ensure their on-going safety and integrity. The AMP encompasses a multi-activity approach that includes inspection, recording, trending, degradation prediction, evaluation, judgement and remediation.

8.31. The PCPVs constitute massive civil structures whose performance and duty is largely passive and predicated upon maintaining an adequate margin against failure for all credible loading scenarios including faults and hazards. The structures themselves comprise of significant components that combine to provide the overall robustness and strength of the PCPV. Failure of these components including due to ageing has the potential to reduce the safety margin. The scope of the PCPV AMP is therefore informed on this basis to identify the structurally significant components and their degradation mechanisms. BS4975 also underwrites this approach and recommends specific surveillance activities that support the on-going evaluation of the structure and any deterioration including observation of deflections, cracking, temperatures and cracking – to be periodically monitored for compliance with the design basis assumptions.

8.32. The scope of the AMP in this section covers the following key aspects, where appropriate:

- Pre-stressed concrete vessel walls, base and top cap.
- Mass/reinforced concrete foundation disc / walls and stressing galleries including water stops, joint fillers and sealants.
- Pre-stressing tendons, ducts and anchorages.
- Internal Insulation.
- Pressure vessel cooling system pipework inside the vessel.
- Monitoring devices embedded in the concrete.
- Rubber bearing pads supporting the vessel.

8.33. These key elements are hereafter used as the specified components of the PCPVs to form a ‘golden thread’ of examples. In the subsequent discussion herein, the bounding case PCPV from the AGR fleet in terms of AMP risk will be presented, as appropriate, for each specified example element above.

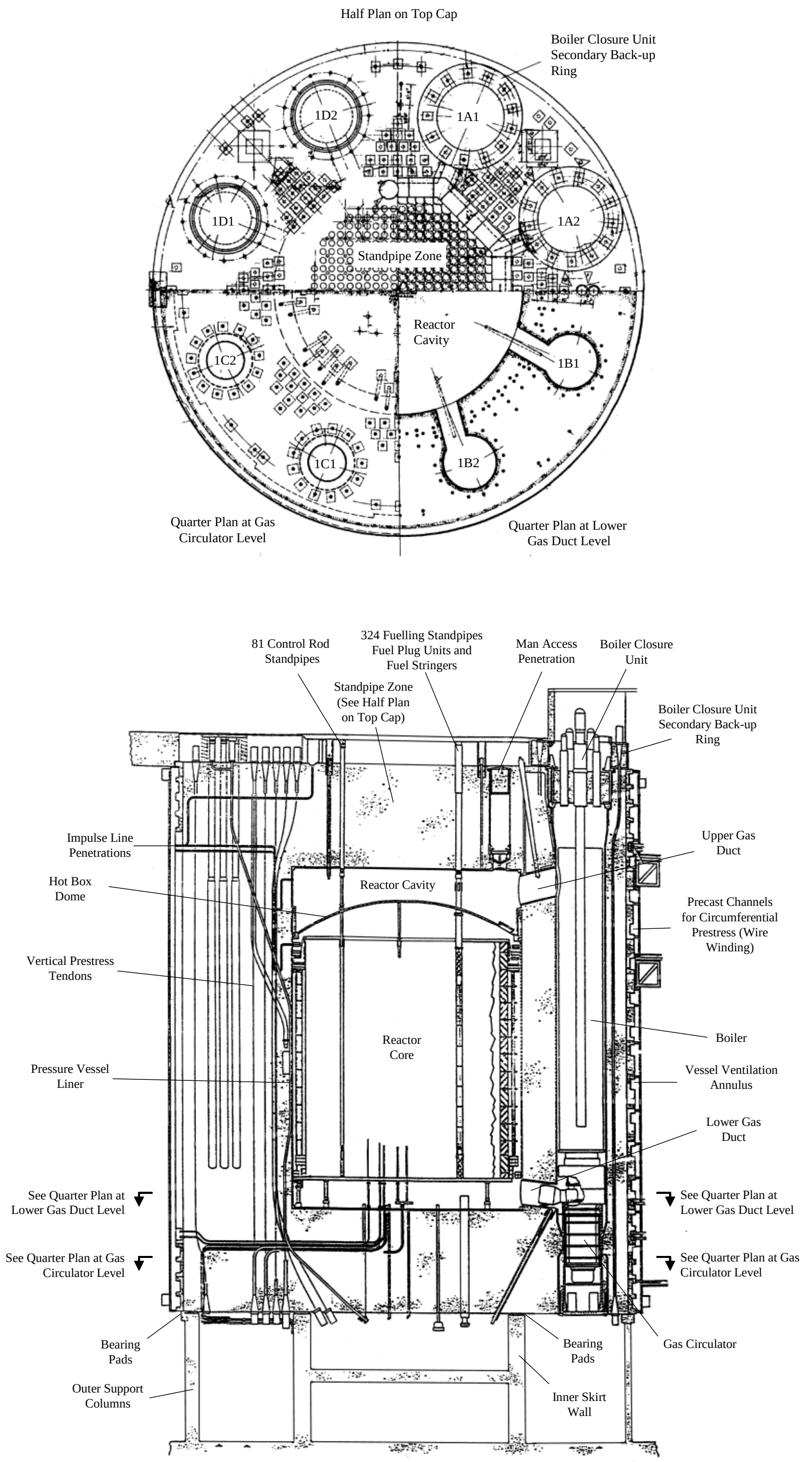


Figure 8.2 Sectional views of the Hartlepool/Heysham 1 PCPV

Processes/Procedures for the identification of ageing mechanisms

8.34. A key element of the AMP is to undertake condition monitoring and inspection of the PCPVs at frequencies and using methods that are appropriate to the importance of the structure to nuclear safety.

8.35. With regard to the condition monitoring of the PCPVs, a distinction may be drawn between this specialised form of surveillance and other types of structural survey. It is therefore useful to define these as follows:-

- Condition monitoring: In-service surveillance of structures on a continuous or quasi-continuous basis using embedded or externally applied monitoring devices. This includes automated systems linked to data-logging equipment which may have pre-defined alarm levels.
- In-service inspection: In-service surveillance of structures by periodic surveys at specified intervals. This may include consideration of information obtained during condition monitoring surveys over the period between inspections.

8.36. In-service inspection of the PCPV and other nuclear safety-related civil structures is undertaken in compliance with licence condition for Examination, Inspection, Maintenance and Testing.

8.37. The minimum surveillance requirements for the PCPV based upon the guidance of BS 4975 consist of the following activities:-

- Concrete surface examinations to monitor and assess the significance of any cracking and the general condition of the concrete.
- Tendon load checks to measure residual force at the anchorage.
- Tendon anchorage examinations to detect signs of deterioration.
- Tendon corrosion examinations to detect signs of deterioration.
- Foundation settlement surveys.

8.38. EDF-NG undertakes the following additional activities to provide further supporting information:-

- Survey of readings of vibrating wire strain gauges (VWSGs) embedded in the concrete and their correlation with theoretical predictions.
- Survey of vessel concrete and liner temperature readings and their compliance with the operating rules for the vessel.
- Main reactor coolant leakage summaries.
- Review of Pressure Vessel Cooling Water (PVCW) leakage events and repairs.
- PCPV deflection surveys.
- Review of operating history for the period under review.

8.39. In the procedure adopted by EDF-NG for compliance against the licence condition for EIMT, PCPV surveillances are led by an Appointed Examiner (APEX) who is a nominated suitably qualified and experienced (SQEP) chartered civil or structural engineer and is responsible for the implementation of the monitoring programme and the assessment and reporting of the results which culminate in a final report.

8.40. As a key requirement of the role, the APEX provides a SQEP surveillance and inspection capability for the PCPV, that is independent of the reactor operation and associated safety case justification.

8.41. The requirement for this level of independence arises from the following:-

- To submit an autonomous SQEP report (independent of station and the corporate "Design Authority") on the PCPV detailing the results and findings from the various activities specified in the station surveillance programme - (akin to an independent technical audit of the PCPV integrity against pre-determined safety case requirements for continued operation).
- To provide an independent oversight role, in recognition of the structural importance and unique form of the PCPV design and construction in relation to its nuclear safety significance.

8.42. As the APEX is a direct employee of EDF-NG, there has to be a suitable degree of independence from the functions of the company organisation that relate to the operation of the PCPVs under their surveillance jurisdiction i.e. the APEX is precluded from fulfilling the Responsible Designer function as defined within INSAG-19.

8.43. The Appointed Examiner has a high level of civil engineering technical expertise, particularly with regard to the design, construction and ongoing behaviour of large concrete structures such as the AGR fleet PCPVs and PWR containment vessels.

8.44. Experience of structural monitoring augmented by specific on-the-job PCPV training plus plant familiarisation is essential covering such topics as:-

- Overview of AGR design & operation.
- Design philosophy of the AGR PCPV.
- Familiarisation with the conditions of the Nuclear Site Licence.
- Familiarisation with station technical surveillance programmes.
- Station safety case.
- Case history and lifetime records.
- Quality Assurance procedures.
- BS 4975 Design Code.
- Safety training (including pre-stressing supervision).
- Plant familiarisation.

8.45. All work related to the structural monitoring of PCPVs is carried out under an approved Quality Assurance Programme (QAP) which is subject to regular review. The QAP provides for the delegation of specific duties of the Appointed Examiner to personnel who meet appropriate training requirements. APEX is the collective term for the personnel tasked with the execution of in-service inspection activities relating to PCPVs as detailed in the QAP.

8.1.2 Ageing assessment of PVPVs

8.46. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

8.47. The ageing assessment strategy has been developed based upon surveillance campaigns that focus upon potential issues at greater risk of ageing degradation. In addition, the recourse to OPEX provides insight into other unforeseen risks that could manifest with time.

8.48. The AMP for the PCPVs for each of the NAR examples is now discussed. The key ageing mechanism for the PCPVs is material degradation / deterioration due to ageing and/or the long term exposure to operating conditions. The AMP monitors the performance of the PCPVs such that an early indication of an ageing related mechanism that could affect the structural performance and reduce the overall design or safety margin is detected. These can be considered for the main structural components:-

- Ageing of the concrete including effects from long term operation (exposure to stresses/strains associated with gas pressure and high temperature effects).
- Surface impregnation of the concrete due to carbonation/chlorides.
- Degradation of the concrete constituents e.g. alkali-silica reaction.
- Concrete shrinkage & continued hydration.
- Thermal transient creep of the concrete.
- Irradiation effects.
- Pre-stressing system – strand relaxation, loss of preload, material changes, grease durability, wear and tear of anchorage components.
- Bearing pads – long term ageing, performance.
- Effect of water ingress and CO_2 leaks on the structural concrete and pre-stressing leading to grease emulsification, corrosion of pre-stressing, stress corrosion cracking.
- Foundation support – long term settlement, tilt, thaumasite attack on buried concrete.
- Thermal insulation.

(i) <u>Pre-stressed concrete vessel walls, base and top cap</u>

8.49. The accessible surfaces of the PCPV are closely inspected to monitor and assess the significance of any cracking and the general condition of the concrete including the presence of any water leakage from the vessel cooling system. The results of these surveys are then recorded and a comparison made with previous surveys.

8.50. Record drawings are maintained that provide lifetime documents of the PCPV surface condition. Details routinely recorded include:-

- Cracks and other visually obvious defects (spalling, delamination).
- Signs of wetting (e.g. PVCW leaks), and any evidence of other significant surface staining.
- Significant areas where access for examination was not possible.
- The operating conditions of the vessel at the time of the examination.
- Concrete surface temperatures recorded during the examination (where appropriate).

8.51. The presence of water leakage is treated seriously as it could be an indication that the integrity of the pressure vessel cooling water system (PVCW) has been compromised. If any signs of leakage are detected, the leakage rate is recorded (if possible) and an investigation carried out to determine the source of the water and the extent of remedial action to be taken.

8.52. Fine cracking and surface crazing has been noted at the surface of the PCPVs due to drying shrinkage and thermal effects on the outer layer of the concrete. This tends to be exhibited at discontinuities and around embedments or associated with construction joints.

8.53. In the event that cracks show unstable behaviour, or are located in a critical position or show an unusual pattern then they are initially assessed by the Appointed Examiner. The use of crack monitoring devices (e.g. demountable mechanical strain gauge) is adopted to monitor long term behaviour. A significant crack is generally defined as a surface crack ≥ 1mm in width and/or a crack which by its location, extent, or relationship to any other defect or feature may be an indication of a departure from the predicted behaviour of the PCPV. If the APEX concludes that further evaluation is required then this is assigned to the PCPV Design SQEP for technical evaluation and authoritative advice.

8.54. The surface examinations are controlled by PCPV specific Work Instructions (WI) that may include additional local examination requirements unique to the structural design, layout or previous operating history of the PCPV. The WIs stipulate that the information recorded shall include dimensions and relationships to nearest physical features on the PCPV as appropriate. If appropriate, cracks and other features shall be recorded photographically with records being clearly identified and stored in the task file. Following the examination, all details shall be transferred to an updated version of the record drawing from the field copy.

8.55. Where access affords the opportunity, the top cap in the vicinity of the congested standpipe zone is visually inspected under the surveillance regime. In addition, a precise optical level survey is performed to measure the deflection of the top cap region of the vessel structure under depressurised and normal operating conditions. The results are compared with measurements taken at the time of the original proof pressure test (PPT) performed during PCPV commissioning. The survey is used to confirm that the PCPV top cap behaves in a predictable linear elastic manner consistent with the results obtained at the PPT.

(ii) <u>Mass/reinforced concrete foundation disc / walls and stressing galleries including water stops, joint fillers and sealants</u>

8.56. The concrete condition of the visible face of the lower foundation disc (or circular support walls as appropriate to PCPV design layout) and stressing galleries (only present at Hunterston B, Hinkley Point B, Heysham 2 and Torness) are included as part of the overall remit of the surface examinations. Evidence of cracking, defects or staining is recorded in a similar manner to the main PCPV structure by the APEX.

8.57. Precise optical levelling is carried out to determine the degree of foundation settlement experienced by the structure and identify any differential settlement of the foundation support which may affect the satisfactory operation of the nuclear related plant and services.

8.58. The type of foundation for each PCPV varies with the type of vessel design, layout and local ground conditions at each site. The tilt of the PCPV foundations is estimated from the settlement results and a comparison is made against safety case tilt limits which are based on an assessment of the effects on the core and its support structure and the ability to safely insert the control rods.

8.59. The lower stressing gallery structure is inspected and any evidence of ground water ingress is noted, which may be indicative of water bar degradation across construction joints or failure or joint seals. The APEX records appropriate recommendations to best manage or address significant instances of groundwater ingress that could threaten the notionally benign atmosphere in the vicinity of the PCPV soffit. In the event that major repairs are required, these would be led by the PCPV Design SQEP(s) subject to APEX agreement.

(iii) <u>Pre-stressing tendons, ducts and anchorages</u>

8.60. A variety of different types of pre-stressing system are used on the PCPVs. However it should be noted that, unlike most other civil structures, all the systems used on AGR PCPVs employ un-grouted tendons to allow withdrawal and in-service inspection and re-stressing of tendons should this be required.

8.61. Corrosion protection of the tendons is provided by various treatments applied to the pre-stressing wire, strand, tendons and ducts. Generally a factory applied corrosion protection system on the wire or strand was augmented by the application of suitable grease during installation of the tendons. This method of protection has performed well in service with tests confirming that effective protection is still provided after 40 years operation.

8.62. The vessels at Hunterston B, Hinkley Point B, Heysham 2 and Torness are all essentially similar in design with helical pre-stressing in both rotational orientations around the structure. The main steam/feed penetrations are located radially around the vessel walls and are located within sleeved ducts in the concrete structure.

8.63. Heysham 1 and Hartlepool are podded boiler designs comprising of vertical and circumferential pre-stressing systems (known as wire windings). The boiler units are accommodated within dedicated cavity ‘pods’ in the PCPV barrel wall rather than being located internal to the structure as per other AGR designs). Hence the main penetrations pass vertically through the top and bottom slab of the structure. The Boiler Closure Units (BCUs) that close the upper penetration for the boiler and feed/steam pipework penetrations are also pre-stressed in a circumferential wire windings manner.

8.64. The Dungeness B vessels incorporate a pre-stressing system of vertical and circumferential high tensile steel tendons contained in ducts within the concrete and externally via ‘blister’ end anchorages. The pre-stressing system is the BBRV button-head system (unique to Dungeness B). Each vessel comprises vertical tendons, circumferential barrel tendons and circumferential tendons in each of the top and bottom slabs.

8.65. The AMP for the pre-stressing systems is primarily informed from dedicated surveillance as follows:-

- Tendon load checks (or equivalent to measure residual levels of pre-stressing).
- Pre-stressing strand removal for inspection and mechanical testing.
- Anchorage inspections and refurbishment.
- Grease sampling and monitoring.

8.66. The residual PCPV tendon loads in the pre-stressing system are assessed for loss of load by performing ‘lift-off’ operations at the tendon anchorages in a sample generally consisting of a minimum of 1% of all tendons. The load checks are tabulated and presented in graphical form by the APEX so that trends can be compared with design expectations and the maximum permissible limits of loss of load.

8.67. The level of pre-stress reduces naturally with time due to concrete shrinkage, creep and tendon relaxation so the results from sample load checks are used to monitor trends which can be extrapolated to end of life by plotting them against time in a log-linear format. By this means, an early warning is given indicating that the PCPV mean tendon loads are approaching the minimum values required by the design and remedial action can be taken to restore the pre-stress in the vessels or substantiate the longer term safety case limits.

8.68. At Heysham 1 and Hartlepool, the circumferential wire-winding pre-stress is primarily assessed by reviewing and trending the output from Change of Load Indicators (load cells) located in-board of the band of pre-stressing at specific recessed monitoring points in the vessel outer wall.

8.69. For each PCPV a number of lengths of pre-stressing strand are withdrawn for metallurgical and mechanical testing and comparison with the original specification. The withdrawn strand is closely inspected for evidence of corrosion or other defects. Corrosion may result from wetting due to leakage of PVCW pipes cast within the vessel concrete or from external sources (for example from ground water seepage or spillages, plant leaks). Alternatively, corrosion could occur due to the hydrogen embrittlement or from the presence of microbial induced corrosion from bacterial contamination of the strand present before tendon placement. The major cause of corrosion has been found to be absence of full grease coverage on key components or localised lengths of pre-stressing arising from original construction or where the strand surface protection has been adversely affected by the presence of water leakage leading to grease emulsification.

8.70. The anchorage components are inspected for signs of deterioration and new grease protection is applied to maintain corrosion protection. Any blockage of drain holes on the anchor bearing plate is cleared to allow free drainage as required in the event of PVCW leakage and to prevent accumulation of grease oil in the base of the tendon duct.

8.71. Additional sampling and testing are also carried out on the corrosion protection medium for the tendons and their anchorage components to confirm that adequate protection is provided throughout the life of the PCPVs.

(iv) <u>Internal insulation</u>

8.72. The inside surface of the PCPV concrete is faced with a steel liner anchored into the concrete structure with propriety layers of thermal insulation on the reactor internals side. The steel liner forms a pressure boundary component that is welded to penetration sleeves as they pass through the PCPV walls (or closure unit as appropriate to PCPV design). The internal thermal insulation protects the steel liner, penetration sleeves and the surrounding bulk concrete from the worst effects of the internal PCPV operating temperatures and in combination with the PVCW system maintains the critical components with an acceptable temperature range. Ageing mechanisms considered include:-

- Material deterioration/ prolonged exposure to high temperature.
- Loss of seal integrity between insulation panels.
- Loss of support / fixity.

- Volume changes in the insulation material.

8.73. For the podded boiler design PCPVs, the podded cavities are also finished with a steel liner and propriety insulation to mitigate the effects of the reactor gas temperatures in the vicinity of the boiler units. The closure units comprise wire wound pre-stressed concrete plug units that close the top end of the podded cavities.

8.74. Thermocouples are used to check that concrete and steel liner temperatures are maintained within the limits specified in the operating rules for the vessels and that there is no upward drift of temperature with time or excessively hot areas within the structure due to, for example, failure of the liner insulation. Acceptance criteria for operating temperatures are specified for each PCPV in station operating instructions.

(v) Pressure vessel cooling system pipework inside the vessel

8.75. Supply of PVCW is essential to maintain PCPV cooling and ensure that the concrete temperatures remain within acceptable limits. A loss of PVCW flow could potentially result in overheating and structurally significant cracking. In addition, PVCW leaks have the potential to cause localised degradation of constitutional parts of the vessel including the tendons, concrete, steel liner, penetrations, gas sampling (indication) lines and embedded PCVW pipework via corrosion that could be more aggressive in the presence of coincident CO_2 leakage from pressure boundary components.

8.76. All of the PCPVs carry risks or varying probability and consequences, associated with water leaks and/or significant loss of vessel cooling water flow. A common problem at some PCPVs is leakage from inaccessible sections of the embedded PVCW pipework within the vessel. Historically, the older vessel designs have exhibited a greater resilience to PVCW related leaks. Ageing mechanisms for the PVCW pipework that are considered under the AMP include:-

- Pipewall corrosion (manged by PVCW water chemistry control).
- CO_2 leakage into the PVCW system (reducing the pH level).
- Fatigue cycling of pipe welds from operating pressure/temperature conditions.
- Vessel movements (micro-movements due to internal vessel stresses/strains) – particularly during start up / shutdown transitions.
- Water hammer effects from pump transients.
- Long term ageing, wear and tear of the steel pipework and connections.

8.77. The AMP objective is informed by surveillance to determine whether there have been any water leaks, including from the PVCW system, that may have affected the PCPV and, if so, to assess the significance for the integrity of the PCPV.

8.78. The surveillance includes the following key activities:-

- Routine walk downs of the PCPV looking for signs of water leakage.
- Routine inspection of tell-tale drains and sumps.
- Longer term trends reviewed by the PVCW Leak Assessor and the APEX.
- Assessing water leaks reported in accordance with the company technical standard for assessment of PVCW and CO_2 leaks affecting the PCPVs.

8.79. A fleet wide database has been established that provides a record of all known leaks over the operating history of each vessel, including details of leak location, date, leak rate, identification of affected piping system and record of repairs. This database is used for trending and to support condition monitoring.

(vi) Monitoring devices embedded in the concrete

8.80. Embedded instrumentation is used to monitor the vessels in the form of vibrating wire strain gauges (VWSGs). It should be noted, however, that in the majority of PCPVs the VWSGs were installed for intended use only during the commissioning Proof Pressure Test and were not intended for long term monitoring although they have subsequently been used for this purpose. There are also some VWSGs that provide concrete temperature measurements that inform the station operating instructions for compliance control.

8.81. The results of the strain gauge measurements are reviewed periodically and fully analysed at five yearly intervals for comparison with theoretical predictions. They provide a useful indication of elastic behaviour by the concrete during pressure cycles and long term creep effects but the results must be considered as being advisory only. To date the results of the Vibrating Wire Strain Gauge measurements have shown a reasonable correlation with expected values.

8.82. Thermocouples provide data on the maximum bulk concrete temperatures to ensure they are maintained within limits (this helps to ensure that the strength of the concrete is unaffected and that creep strain is minimised). The liner/concrete temperatures are maintained by the PVCW pipework which is attached to the concrete side of the steel liner close to the barrel liner and around penetrations.

8.83. In all AGR PCPVs the temperature differential between the concrete in the standpipe region and that in the surrounding concrete is limited to prevent differential expansion/contraction leading to a reduction in the compressive pre-stress in the standpipe region.

8.84. The value of the Vibrating Wire Strain Gauge output for PCPV integrity is clear from an AMP perspective. Recent EDF-NG initiatives have focused mainly on addressing the obsolescence of monitoring devices attached to steel components within the PCPV as these largely comprise the reactor pressure boundary components. A fleet database of strain and temperature measuring instrumentation has been developed and initiatives to address obsolescence have been undertaken to refurbishment instruments that were previously declared unavailable. Under the on-going remit of the PCPV Through Life Management Strategy, it is proposed that the concrete instrumentation is included within this database to raise its profile in terms of adding value to the AMP and also to ensure that SQEP Instrumentation & Control expertise and oversight can be sought to improve the longer term functionality of the devices.

(vii) Rubber bearing pads supporting the vessel

8.85. The loads from each PCPV are generally transmitted to its foundation by neoprene rubber bearing pads. The integrity of the bearing pads continue to be assessed and findings to date (from testing, OPEX and reference to published literature) have concluded insignificant degradation of the material properties of neoprene and natural rubber.

8.86. For the operating conditions applicable to the PCPV bearing pads and the generally benign passive environment (i.e. not exposed to UV or atmosphere) it is judged that the PCPV bearings are not exposed to a high risk of degradation. Irradiation effects are not considered to be significant as bearings are located outside of the PCPV structure which acts as a radiation shield to prevent external radiation effects. Furthermore, the internal steel plates present in the elastomeric bearings are not exposed to the environment and no specific concerns have been identified from the reviews to date.

8.87. The main causes of degradation to both natural rubber and chloroprene rubber are oxidation, attack by ozone or degradation by UV radiation. The benign conditions of exposure of the PCPV bearings present a low risk of degradation due to these mechanisms.

8.88. Rubber materials can be prone to oxidative degradation through reaction of unsaturated chemical bonds in the material with oxygen or ozone. This is more pronounced for natural rubber materials than for neoprene materials. Oxidation causes the surface hardness to increase, which can lead to cracking under load. Commercial bearing products (akin to those used on the PCPV fleet) would be expected to contain anti-oxidation additives.

8.89. The consideration of fault conditions which are specific to the AGR environment on the degradation of the two types of elastomer have been assessed. These fault conditions considered include exposure to fluids and exposure to steam or CO_2 at high temperatures.

8.90. In the case of high temperature exposure to steam or CO_2, deterioration of either material is expected at temperatures in excess of 100°C. It is expected that the measures in place to manage the release of high temperature gas are sufficient to mitigate the risk of bearing material deterioration due to exposure to high temperatures.

8.91. The normally visible outermost face of the PCPV bearing pads can be inspected by the APEX during the surveillance of the concrete surface condition in the vicinity of the main foundation disc. For those PCPV support layouts that comprise circumferential walls the access to inspect is restricted by cooling pipework and hot gas release insulation and specific inspection operations have to be established to allow visual examination.

8.92. Samples have been removed from some PCPV bearing pads at some AGR sites during past surveillances (>20 years ago) and found no evidence of significant degradation. Further sampling is currently under ALARP consideration by individual APEXs on a station-by-station basis. At present there are no formal requirements on the APEX to visually examine the accessible faces of the outermost bearings but a greater emphasis is now being placed upon including them within the overall surveillance remit.

8.93. In terms of acceptance criteria, the bearing pads provide a structural role to provide support to the PCPVs and transmit loads to the foundation structure including the lateral effects of the pre-stressing. Tilt measurements of the vessels provide a measurement of any differentiation in the vessel verticality (that could be indicative of the bearing degradation). However, the acceptance is largely judgement based and informed by the absence of step changes in the bearing environment, the continued passive appearance of the bearing compressed under the PCPV loads and recourse to OPEX. The AMP would benefit from broader utility feedback and experience on long term rubber bearing pad performance beneath nuclear structures (see paragraph 8.219).

Key design documents, standards and guidance used to inform the AMP

8.94. A key part of the ageing management process includes an assessment of significant changes to relevant standards, principles and methodologies since the time of original construction.

8.95. The concrete design codes at the time of the early PCPV development were CP 114 and CP 115. Both these codes were replaced by a unified code CP 110, which was later superseded by BS 8110 in 1985.

8.96. The pre-stressing strand for the PCPVs was supplied in accordance with the manufacturer's own specification, 'British Ropes Ltd Specification CS31/69'. The current standard for pre-stressing strand is BS5896:2016.

8.97. The current design standard for PCPVs is BS4975:1990. When first issued in 1973, this code incorporated the procedures used and experience gained from the design of all of the previous Magnox and AGR PCPVs.

8.98. A detailed review of the design and construction of the PCPVs against these more recent standards was included within the scope of the Periodic Safety Reviews (PSR) across the fleet. No significant deficiencies were identified and there have been no further changes to BS4975.

8.99. The IAEA Safety Guide NS-G-2.12 Ageing Management for Nuclear Power Plants (Ref 17) sets down the principles for managing nuclear safety structures to ensure the availability of required safety functions throughout the service life of the plant, with account taken of changes that occur with time and use.

8.100. This requires addressing both physical ageing of structures, systems and components (SSCs), resulting in degradation of their performance characteristics, and obsolescence , i.e. their becoming out of date in comparison with current knowledge, standards and regulations, and technology.

8.101. A foundation for effective ageing management is that ageing is properly taken into account at each stage of a plant's lifetime, i.e. in design, construction, commissioning, operation (including long term operation and extended shutdown) and decommissioning. The effective management of PCPV ageing is a key element of the safe and reliable operation of nuclear power plants.

8.102. The primary objective of the Safety Guide NG-S-G-2.12 is to provide recommendations for managing ageing of SSCs important to safety in nuclear power plants, including recommendations on key elements of effective ageing management. As a nuclear licensee operator EDF-NG complies with the principles which support the establishing, implementing and improving of the systematic ageing management programmes for the PCPVs. The EDF-NG approach aligns with the Safety Guide and seeks to focus on managing the physical ageing of the PCPVs and components important to safety. The Guide also provides recommendations on the safety aspects of managing obsolescence and on the application of ageing management for long term operation.

8.103. IAEA Safety Report Series No.82 (Ref 54) and NP-T-3.5 (Ref 55) describe practices and techniques for the inspection, mitigation of ageing degradation, corrective action including repair methods, and operating experience for concrete containments. It also provides general guidance for developing an effective ageing management process for concrete structures.

8.104. IAEA also produce a programme of International Generic Ageing Lessons Learned (IGALL) reports for Nuclear Power Plants. These reports supplement the IAEA Safety Reports Series No. 82, Ageing Management for Nuclear Power Plants. For structures, systems and components important for safety, the following information is presented:

- a generic sample of ageing management review tables;
- a collection of proven ageing management programmes; and,

- a collection of typical time limited ageing analyses.

8.105. One IGALL document that is relevant to the ageing management of the concrete containments that can reasonably be read across for PCPV interpretation is listed below:

- AMP 302 ISI for Concrete Containment.

8.106. This safety report describes practices and techniques for the inspection, mitigation of ageing degradation, corrective action including repair methods, and operating experience for concrete containments. It also provides general guidance for developing an effective ageing management process for concrete containments based upon the following principles.

- Understanding ageing mechanisms.
- Detection of ageing effects.
- Monitoring and trending of ageing effects.
- Mitigating ageing effects.
- Acceptance criteria.
- Corrective Actions.

8.107. The IAEA guidance provides specific information for in-service inspection (ISI) for managing ageing of containment reinforced concrete including un-bonded post-tensioning systems. Whilst the IAEA guidance is primarily intended for PWR containment type structures (rather than AGR PCPVs) it can be cross-read over to allow sensible interpretation for the PCPV structures. In this context it is judged that the EDF-NG procedures for ageing management adequately capture and address the aforementioned key IAEA principles and there are no gaps identified.

Research and Development

8.108. Historically, a number of research projects have been commissioned. Many of these initiatives were primarily relevant to AGR PCPVs. The general areas of study that have been considered in these research projects are summarised below:

- Effects of elevated temperatures: Assessment of structural integrity of concrete at elevated temperatures.
- Concrete properties: General in-situ properties of concrete in safety related structures. Effects of cycles of elevated temperatures/cooling on restrained concrete. Ageing/durability at elevated temperatures.
- Behaviour of concrete at elevated temperatures, including multi-axial loading and development of constitutive models.
- Fracture energy at elevated temperatures.
- Inspection and maintenance: NDT methods for thick concrete sections. Techniques for locating and sealing PVCW leaks.
- Survey methods: Review of advanced survey techniques and their application.
- Analysis methods: Modelling the effects of gas in cracks. Use of finite element modelling in linear and non-linear analysis (thermal and ageing processes were not addressed).

- Tendon corrosion/protection: Long-term performance of greases used for tendon protection. Extent of corrosion in unprotected tendons.
- Tendon loads: Measurement of tendon loads. Tendon load profile modelling. Instrumentation for monitoring of tendon loads.

8.109. The research studies to date have broadly supported and enhanced understanding of a wide range of potential ageing and degradation issues. In particular, the investigations into the performance of grease for protection of tendons, and the extent of corrosion in unprotected tendons have provided informative results. These results are of practical use in justifying the long term integrity of pre-stressing tendons. The general conclusions to date confirm that the protection provided by the grease remains effective.

8.110. There are a number of projects that have considered the structural integrity of concrete at elevated temperatures, together with the properties of concrete at both normal ambient and elevated temperatures. Some of these projects have also considered ageing and durability issues related to vessel concretes. The findings of some of these studies have been incorporated into finite element analysis work, and structural integrity assessments, that have been performed for the PCPV under fault conditions.

8.111. It is also noteworthy that the multi-axial testing of concrete at elevated temperatures is still ongoing, and it is anticipated that the results of these studies will provide data for the further refinement of concrete constitutive models in future analysis work.

Operational experience (OPEX)

8.112. As a fundamental part of the AMP, operational experience and feedback is routinely reviewed to ensure that any issues of relevance to the PCPVs are assessed to allow any lessons learned to be incorporated into the ageing management strategy.

8.113. The UK nuclear industry is the only country to operate AGRs and hence most of the operational experience is internally based upon observed plant performance or via safety reviews to assess ageing risks.

8.114. By careful concrete mix design and quality of construction, the PCPV structures have not exhibited any adverse evidence of internal chemical reaction leading to concrete material degradation or incompatibility with the immediate operating environment. This includes the absence of any adverse indications relating to Alkali-Silica Reaction (ASR), delayed ettringite formation, carbonation, chloride attack or reduction in strength. The following AMP risks are highlighted:

(i) Thaumasite Sulphate Attack (TSA)

8.115. A thaumasite form of sulphate attack (TSA) is a very uncommon reaction, which is only known to occur in buried concrete where all four conditions of very high sulphate levels, plentiful moisture, a supply of carbonate (usually from the aggregate in the concrete e.g. limestone) and low temperatures (below 15°C) are present. When it does occur, its consequences can be severe.

8.116. The PCPVs and support structures are typically located in an enclosed environment, and, although there is occasional wetting of the concrete surface, the vessel concrete is not in sufficiently wet conditions, or sufficiently low temperatures, for this type of reaction to occur.

8.117. The supporting foundations are in contact with groundwater, and the temperature around the foundations will generally tend to be below 15°C. A very small number of cases of TSA have occurred in buried concrete associated with highway structures containing siliceous aggregates, suggesting that concrete containing little or no carbonate can be affected if an external carbonate source is available. The best available guidance for the use of concrete in aggressive ground is provided in BRE Special Digest 1. This was first published in 2001 and was most recently updated in 2005. The latest edition consolidates the guidance provided in previous editions and takes account of recent research findings on TSA. The risk of TSA affecting the PCPV foundations has been assessed and is judged to be low and any deterioration would be detectable from the settlement surveys.

(ii) Irradiation

8.118. The effects of irradiation of the PCPV concrete can include swelling of the aggregate and the generation of pressures in the cement paste. It is considered that irradiation can cause heating of the concrete, although this is not expected to be significant compared with the effects of other sources of heating. Fast neutron doses could also cause reductions in concrete strength, although this was not expected to be significant.

8.119. Reference can be made to guidance on the effects of irradiation of concrete presented in the PCPV design code, BS 4975: 1990, which states that irradiation effects will be insignificant at fast neutron doses of less than $5x10^{21}$neutron/m². The maximum total radiation exposure for an AGR PCPV over an estimated period of 30 years operation at a load factor of say 85% can be shown to be:-

- Gamma radiation 1011rad (109Gray).
- Thermal neutrons $6x10^{23}$neutron/m².
- Fast neutrons $2.3x10^{22}$neutron/m².

8.120. These doses are considered to be sufficient to cause some damage to the PCPV concrete, but this would be expected to be limited to, say, the inner 200mm of the wall due to attenuation effects. The effects on the integrity of the PCPV are therefore expected to be insignificant, and it was noted that the same conclusion had been reached in a review of the radiation effects on PCPV concretes over successive Periodic Safety Reviews (PSR) assessments.

8.121. The longer term AMP would benefit from wider utility experience and feedback on the effects of long term irradiation exposure of structure concrete to underwrite the current understanding and assessment of lifetime risk to the PCPV integrity (see paragraph 8.215).

(iii) Tendon and tendon anchorage component corrosion

8.122. The tendon wires were coated with phosphate during manufacture and spun into strands which were coated with a surface protection (e.g. at Heysham 1 and Hartlepool this was Castrol Rustillo DW 932) before being transported to site. During tendon installation, the strands were pulled through a stuffing box which applied a layer of grease protection (e.g. at Heysham 1 and Hartlepool this was Castrol S202). The ducts were not generally sealed and the anchor components were coated with Shell Ensis fluid or similar. At Heysham 1 and Hartlepool, the outer circumferential wire windings were coated with lower grade viscosity grease (Castrol S203) during winding to fill all the interstices between the wires in the channels.

8.123. Routine inspections have been carried out for corrosion during construction and operational periods. Pre-stressing strands that have exhibited evidence of corrosion or local wire failures (within the twisted strand) have been replaced (in main pre-stressing systems). This has led to a wider scope of tendon examination to assess the condition of adjacent tendons that may have been subject to similar wetting action. A recorded database of wetted tendons is maintained for each PCPV to assist the choice of tendons to be examined at each surveillance campaign. The present AMP arrangements for pre-stressing surveillance and monitoring are judged adequate to detect any step change in material condition.

(iv) Concrete creep

8.124. Under the action of applied stress, concrete undergoes a significant creep strain, even at ambient temperature, although the rate of creep increases markedly with temperature. In the PCPVs, the concrete is held in a state of compressive stress via the post-tensioned pre-stressing tendons. Concrete creep is therefore a contributing factor to the gradual reduction in tendon load and needs to be taken into account in any long term assessment. Concrete creep may be regarded as proportional to stress and this allows the use of the parameter specific creep which is defined as the creep strain per unit stress (ε/MPa); this parameter simplifies the numerical creep analyses of PCPVs.

8.125. A number of studies have been carried out on the long term deformation behaviour and other physical properties of the PCPVs. For example, experimental creep data for the effects of temperatures (20^{o}C, 65^{o}C and 95^{o}C) and age at loading, using an applied stress of 13.8MPa, was obtained from samples of the Hartlepool or Heysham 1 concrete mixes. The results were used for the initial creep predictions. It was established from the creep predictions that creep strains were well within expectations. In a more recent study creep predictions have been derived from various predictive models for PCPV concrete structures.

8.126. In the case of the PCPV modelling, concrete creep has been calculated by finite element analyses as part of studies to compare total predicted strain with the strain values measured from the embedded strain gauges. The measured and predicted rates of strain change are generally found to correlate well for most gauge locations, indicating that the concrete creep over the period of review is broadly in line with expectations.

8.1.3 Monitoring, testing, sampling and inspection activities for PCPVs

8.127. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

8.128. In-Service Inspection (ISI) of the PCPVs is controlled by the station MS or MITS as relevant to the particular station arrangements, which specifies the scope and frequency of the PCPV surveillance activities.

8.129. The PCPV surveillance reporting cycle takes places every three years which generally aligns with the statutory outage programme and reporting requirements for ONR regulatory consent to start-up for a further period of three years operation.

8.130. For each of the representative elements identified in Section 8.1.1 the bounding case PCPV from the AGR fleet in terms of AMP risk will be included where appropriate in the following discussion.

(i) Prestressed concrete vessel walls, base and top cap

8.131. Generally, the surveillance of the PCPVs has shown that the concrete surface cracking is confined to drying shrinkage and thermal strains and no instances of significant structural degradation has occurred. Some evidence of surface spalling has been recorded but not of structural significance to the long term integrity.

8.132. The bounding case PCPV structure has experienced localised barrel cracking at mid-height which is an area more vulnerable to the propagation of surface cracking. This is due to the larger movements associated with normal operating pressure conditions and hence this area is examined under both pressurised and depressurised conditions.

8.133. The largest cracks are monitored by demountable mechanical strain gauge surveys at both pressurised and depressurised conditions to assess crack behaviour in response plant operating conditions with time. The results are trended and demonstrate that the crack width reduces between pressurised (at-power) and depressurised (shutdown) conditions. In addition, the long term trend indicates that the cracks are not growing significantly in width at the surface. The cause of the original cracking has been attributed to drying shrinkage action and the notable crack appearance on the surface is emphasised by the location and pressure cycling.

8.134. Forward monitoring of the demountable mechanical strain gauge survey results will continue to monitor for step changes and to support lifetime assessment. The AMP would benefit from utility feedback and experience in the application of continuous monitoring of concrete surface cracks using remote electronic techniques that could improve upon the use of discrete point-in-time demountable mechanical strain gauge readings (see paragraph 8.215).

8.135. The APEX is required to interpret and report on the examination of the concrete surfaces and assess the structural significance of any defects for the PCPV. If any abnormalities are identified, the Station Director and company Design Authority must be informed. A Condition Report is raised for recommendations from the APEX final statutory report to track progress against any proposed repairs or further monitoring. Due to the safety significance of the PCPV, the rectification of defects is generally given a high priority via the station work management arrangements.

8.136. The top cap deflection survey results across the PCPV fleet generally confirm linear elastic behaviour between pressurised and depressurised conditions and show reasonable alignment with the deflections recorded during the PPT commissioning.

(ii) <u>Mass/reinforced concrete foundation disc / walls and stressing galleries including water stops, joint fillers and sealants</u>

8.137. No major defects have been found on the foundation discs or circumferential walls that support the PCPV structures, as appropriate.

8.138. Groundwater ingress to some lower stressing galleries has been observed. The bounding case PCPV has experienced groundwater ponding around the lower stressing gallery floor with the need for pump extraction measures to be implemented. Injection grouting of the outer lower stressing walls has taken place to reduce the permeability against water ingress and refurbishment of the seal joints has been performed.

8.139. The lower stressing galleries and PCPV soffits (depending upon PCPV layout) are inspected routinely at typically 3-6 monthly interval frequencies. Any emergent defects are advised to the APEX for assessment.

8.140. The stability of the foundations under normal and hazard loading are periodically assessed in relation to ground conditions and shown to be satisfactory. The recorded settlement of the fleet PCPVs has been reviewed and generally found to be satisfactory. No credible mechanisms that could cause a sudden failure of the foundations without advance warning have been identified.

8.141. In terms of foundation settlement, only one of the PCPV sites has been affected by ground settlement. Settlement of the large raft foundation supporting both PCPVs and the adjacent Fuel Handling Unit has been continuously monitored since shortly after the top concrete pour of the foundation raft was completed.

8.142. Settlement and tilt surveys based upon precise levelling technique based upon a site permanent datum are generally undertaken every five years. At the bounding site for PCPV foundation settlement, the surveys are undertaken annually to provide enhanced surveillance. The objective is to monitor movements that might adversely affect the integrity of the PCPVs. The main movements to be considered are overall settlement and tilt of each PCPV, and differential settlement between them. In addition, the Reactor Building raft foundation upon which the PCPVs are supported needs to be assessed in the region of the PCPVs. Finally, interaction between the Reactor Building raft and adjacent structures is considered.

8.143. The results from the foundation tilt surveys are assessed against acceptance criteria as follows:-

- 5 minutes of tilt.

  Advise Graphite Project Team to the level of tilt. This may influence the choice of fuel channels for inspections.

  Survey the interface between the CO_2 pipework and the PCPVs which could be significantly affected beyond this level of tilt.

  Review small bore CO_2 lines with small seismic margin.

- 7.5 minutes of tilt.

  Provide additional surveillance of boiler operational performance.

- 10 minutes of tilt.

  Review - this level of tilt is so far removed from current levels that it is likely that additional evidence would be available and more precise criteria could be defined. However, there is no evidence of a cliff edge at this level associated with failure to trip, shut down, hold down, or cool the reactor.

8.144. The measured settlement (circa 30-40mm) at the bounding PCPV site has been generally larger than values that were predicted in the original design, and further movement has continued to take place at a relatively slow rate of settlement. Additional assessments have been carried out during the operational life of the PCPVs which consider the settlement, overall stability and structural integrity of the raft foundation under normal operating conditions, together with hazard loading conditions.

8.145. It is concluded from these additional assessments that the raft remains capable of providing safe and stable support to the PCPVs under normal operation and prescribed hazard loadings.

(iii) <u>Prestressing tendons, ducts and anchorages</u>

8.146. Tendon load checks are typically performed every three years as specified in the station maintenance arrangements to measure the residual Anchorage Loads (AL) in a representative selection of tendons.

8.147. An allowance of 3% on the measured results has historically been made for internal duct and anchorage friction effects to derive a mean Effective Anchorage Load (EAL). However, following an internal study into the method of tendon load measurement in PCPVs, a friction allowance of 5% was proposed for the losses to derive the EAL. Therefore, as part of the AMP this higher value is now adopted for the assessment of tendon load results and applied retrospectively to AL values included in the longer term trending.

8.148. The mean EAL is compared with the Minimum Design Load (MDL) specified in the safety case and plotted with previous mean EAL values. A best-fit regression line is derived which can be used to estimate future residual loads.

8.149. Across the PCPVs, the tendon loads reduce with time based upon shrinkage, creep and relaxation effects. This has remained within design expectations of the anticipated pre-stressing behaviour over the PCPV operational lifetime. With time, the tendon load losses reduce exponentially and reach a lower plateau of only minor continuing losses. This behaviour has been noted on all the PCPVs and aligns with the expected structural performance.

8.150. At some PCPVs, the forecast trend of tendon loads is approaching the MDL. The MDL does not represent a fundamental safety limit for the PCPV integrity and scope would exist in the future to reduce the MDL limit whilst maintaining an adequate structural margin in the vessel.

8.151. For the bounding case PCPV, in terms of tendon load reduction, the predicted residual pre-stress is currently estimated to approach MDL before the end of generation. In the event that the MDL was breached (based upon the test results from a future tendon surveillance), the available margins can support continued operation of the plant in the short term whilst further investigations are undertaken, and thus enable the consequences for the safety case to be considered and addressed within an acceptable time-scale.

8.152. The tendon surveillance also involves the de-tensioning and removal of selected strands for inspection and test. As the strands are removed from the bottom of the vessel, replacement strand is drawn in at the top. The full tendons are then re-tensioned to approved levels.

8.153. The removed strands are cut into lengths of approximately two metres, uniquely tagged, and removed for cleaning and visual inspection. Following inspection, one specimen length is selected from each strand and sent for mechanical testing (breaking load, proof load, elongation and elastic modulus) against the requirements of BS5896 and the original specified minimum strength requirements of British Ropes Specification (original strand design specification).

8.154. The visual inspection of tendon strand has no defined acceptance criteria however any abnormal features are selected for appropriate further examination (either metallurgical or mechanical).

8.155. The results of the metallurgical examination will include measured pit depths. For 18mm diameter vertical tendon strand, studies have shown that occasional pits up to 1mm will not jeopardise tendon integrity. For 5.1mm diameter circumferential tendon wire (e.g. podded PCPV pre-stressing), studies have shown that occasional pits up to 0.5mm will not jeopardise tendon integrity. It should be noted however that given the variation in the nature and frequency of pitting and in loss of cross-section and risk of environmentally-assisted corrosion, it is not possible to identify definitive corrosion criteria to cover all circumstances. Therefore, where evidence indicates a reasonable likelihood that pitting may approach or exceed the nominal 1mm criterion, the tendon is considered for selection for complete replacement taking into account other potential reasons for elevated risk (e.g. condition of other tendons in the vicinity, degraded grease or environmentally assisted corrosion risk factors). Where partial examination of a tendon reveals pitting in excess of 1mm, and full tendon replacement is not undertaken, the proposal to return to service with potentially degraded strands is justified by the safety case MDI process.

8.156. Where corrosion as a result of wetting is identified, the severity of damage is compared with the estimated corrosion rates. These are ~2μm/yr for dry tendons;

~120μm/yr for wet tendons with some grease protection. As grease protection diminishes, corrosion rates increase to 50 to 300μm/yr in low chloride conditions and 400 to 1000μm/yr in high chloride conditions.

8.157. At one of the PCPV sites which is entering its late operational life, the requirement to replace the removed strands has been relaxed on the basis of adequate residual margin remaining in the vessel and absence of adverse indications from previous surveillance. This change also reduces the surveillance activities with associated personnel risk (balanced against the very minor benefit to the PCPV of replacing the strand at this late stage in its lifecycle). This relaxation was justified via the MDI process. The AMP in this instance has reflected the later life requirements of the PCPV maintenance.

8.158. For the same station, the interval period between tendon load surveillances going forward has also been increased from two years to three years for similar reasons. These two examples illustrate how the AMP can change in the later life of the PCPVs to reflect previous operating performance and balance of risk for the remaining generation period. These changes are only implemented where clear safety margin can still be demonstrated.

8.159. A selection of pre-stressing system anchorages is visually examined to check their components and supporting concrete for signs of defects such as corrosion, mechanical damage, cracking, or spalling concrete. Any evidence of significant strand slippage is noted. The inspections typically cover the anchorages associated with the tendons selected for residual load-checking and strand replacement.

8.160. For the podded AGR PCPVs, the pre-stressing design layout allows a minimum of 30 vertical tendons over the three year statutory outage cycle to be selected for remote video probe inspections via the bearing plate breather hole. Tendons selected for probe inspection are targeted based on degradation risk. Inspections may be undertaken during refuelling outages or the statutory outage noting that where results may influence strand withdrawal selection, probe inspection are undertaken early in the operating cycle in order to give time for planning any subsequent strand withdrawals in the next statutory outage. Tendon air flow checks using an anemometer have also been undertaken to assess the influence of air flow within the tendon duct and its capability to maintain a generally dry environment within the ducts. The presence of localised blockage to positive air flow has been encountered due to excess grease plugging the available spaces in the duct but subsequent strand removal and examination on such tendons to date does not indicate any adverse effect on the pre-stressing performance.

8.161. The anchorages are visually inspected during load-checking by the tendon stressing contractor with findings presented in a tendon surveillance report. After completion of the tendon surveillance some of the anchorages are re-inspected by the Appointed Examiner.

8.162. The selected tendon anchorages and supporting concrete inspected are generally seen to be in a satisfactory condition. At some bottom anchorages the components and strand tails exhibit localised minor surface corrosion but this is not significant and is deemed to be adequately monitored by the current inspection regime. Measurement checks for the presence of CO_2 in the vicinity of the PCPV lower anchorages is routinely undertaken before reactor start up from refuelling and statutory outages and periodically around the BCUs in the vicinity of the outer annulus.

(iv) Internal insulation

8.163. Two types of insulation material are employed across the AGR fleet within the reactors comprising of metallic foil insulation or mineral fibre. Whilst insulation is partially visually inspected by ISI, this is recognised as being very limited. The effectiveness of the internal thermal insulation is assessed by monitoring the thermocouple data from the steel liner and penetrations to ensure that the temperature range remains within acceptable limits.

8.164. Based upon operational performance to date, it is believed that the insulation “packet” gaps are getting bigger, which is potentially leading to more hot reactor gas passing into the insulation. This has given rise to an increased heat flux to the main reactor dome structure of the internal reactor at one AGR site. The metallic foil is made from 310 Austenitic stainless steel, which is thought to undergo a high temperature (>500°C) phase transformation and volume change, which could theoretically lead to distortions in the packets.

8.165. Whilst the local dome insulation has shown degradation occurring, the insulation on the walls and roof at the affected PCPV do not seem to have been subject to the same degradation effects. Within the insulation flow paths butterfly seals are installed, which act to break up the flow of gas and avoid significant heat transfer. Any degradation of the main concrete vessel or liner thermal insulation would be detected by the liner thermocouples and temperature rises within the PVCW system, and so this is considered to be of low concern.

(v) Pressure vessel cooling system pipework inside the vessel

8.166. The emergence of PVCW leaks is monitored by routine plant walk downs, inspection of tell-tale drains for evidence of flowing water as well as the detailed surface examinations. Once a leak is established, the details are recorded on a fleet database with appropriate details relating to location, extent of leak, affected system, coincident indications and chemistry analysis results. The leak is then targeted for sealing at the next suitable outage (or on load if the access arrangements and safety case allows). The leak sealing process is therefore a reactive measure implemented in response to as-found plant conditions.

8.167. Trending of the leaks is also undertaken to assess for patterns, in relation to location on the vessel, repeat leakage sites and plant transient conditions. Due to the small movements and typically small defect sizes (e.g. pinhole) in the embedded pipework, it is quite common for leaks that manifest during pressurised operating conditions to close once the reactor is in a shutdown condition and depressurised. If access to undertake leak search and sealing operations is limited to outage opportunities, this adds a further complication to the process.

8.168. The leaks resolution process (this refers to the complete process, from leak identification to sealing and the return of the affected pipe to service) can be difficult and time-consuming. For example, this may require one of the two 50% duty cooling systems to be drained down, and so this can only be done off-load. In some cases, due to temperature requirements of the sealing process, the actual sealing operations can only be undertaken during outages with the reactor in a shutdown condition. The sealing itself can be a length activity (long curing time and multiple attempts are sometimes required). Finding the source of a leak in the first place can also be difficult and may depend on the level of experience of the site team.

8.169. When this work is undertaken during outages, interfaces with other outage activities may require use of already limited resources (drain down and reinstatement of the system, pressure testing permits, access scaffold and lagging removal etc.).

Recent evidence suggests that large PVCW leaks can re-open previously sealed leaks thus emphasising the importance of timely leak resolution.

8.170. In terms of an AMP acceptance criteria for PVCW leaks, the optimum and preferred position is to operate the reactor (or return to service after outage) without any known PVCW leaks. Due to the aforementioned logistical difficulties this position may be more aspirational than definitive. However, a number of improvements have been made to the leaks resolution process (see next section) and success in leak sealing has notably improved in recent years. The risks associated PVCW leaks have been rationalised across the fleet to ensure consistency in terms of risk significance, value of risk, and the development of proportionate mitigations. This is an area where the AMP has improved company focus for the PCPV risks.

(vi) Monitoring devices embedded in the concrete

8.171. PCPV behaviour is inferred from data obtained from routine monitoring of embedded vibrating wire strain gauges (VWSGs). During construction, VWSGs were embedded in the PCPV concrete to record concrete strains and temperatures during the vessel proof pressure test (PPT). The gauges continue to be routinely monitored (from a datum point set just prior to the pre-stressing application).

8.172. Those gauges that still function reliably continue to inform the overall structural behaviour of the PCPVs under operating conditions. The gradual failure of gauges, with time, is expected as their original duty was to function for the relatively short duration of construction and commissioning.

8.173. Generally, there are sufficient functioning gauges distributed throughout the PCPVs to allow the general strain state of the PCPV to be assessed. This situation remains stable and recent improvements have been developed to remotely re-invigorate previously defunct gauges. From the data, it can be inferred that the PCPVs remain in a stable, compressive strain state during normal operation, with most gauges showing either no strain change or a slow increase in compressive strain due to creep of the concrete. Short-term elastic strain changes associated with refuelling outages or other reactor trips are visible on the logged strain histories and are consistent with those observed during the proof pressure test (PPT).

8.174. Bulk concrete temperatures are inferred from the VWSGs and also via dedicated concrete thermocouples. The thermocouple output is monitored for compliance against operating instructions (to maintain the bulk concrete within acceptable temperature ranges) and can also inform of potential loss of local PVCW functionality or degradation of the reactor internal insulation at the liner.

(vii) Rubber bearing pads supporting the vessel

8.175. The rubber bearing pads at the PCPV sites have been assessed with respect to their integrity. Previous long term testing of neoprene and natural rubber bearing pads found little degradation of the properties of either material.

8.176. The potential effect of elevated temperature on the rubber bearings for the vessel support has also been assessed. The conclusions of the studies were that the rubber would not significantly extrude when subjected to a temperature of 100°C for several days, and that, under these conditions, the best estimate drop of the reactor would be limited to less than 0.005" (0.127mm). These findings are principally related to hazard conditions out with of normal operating conditions.

8.177. The findings are, however, indicative of the general robustness of the bearing pads, and hence their continued integrity under conditions of normal operation.

8.178. To ensure that the AMP identifies any unexpected degradation, the APEXs are currently reviewing the need to enhance the level of visual inspection of the accessible bearings by the APEX during the routine surveillance with any notable features, geometric changes or signs of distress recorded for further assessment. In addition, where access is feasible, small representative samples from the outermost portion of the rubber pads may be considered for testing to assess for changes in the material properties. Previous inspections of accessible bearings have generally confirmed their satisfactory condition with only the occasional minor defect noted at localised outer sections of the pad.

8.1.4 Preventive and remedial actions for PCPVs

8.179. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

8.180. The PCPVs are large passive structures and by a combination of conservative design philosophy, robust construction and proof testing they are generally resilient to emergent defects, which are primarily slow in their manifestation and detectable. The ultimate PCPV design is generally governed by the effects of actions, not necessarily associated with normal operation, but rather due to requirements to maintain containment and protect against radiation during postulated fault sequence transients. In addition they have to adequately demonstrate a capability to resist the actions from low frequency events e.g. earthquake, explosions, aircraft impact and fire.

Preventative actions

8.181. Preventative measures are primarily addressed by the regular surveillance activities and monitoring to verify that the PCPV behaves in an elastic and predictable manner under normal operating conditions. The physical surveillance activities confirm the material condition and preventative action is undertaken as appropriate, examples of which are as follows:

(i) <u>Prestressed concrete vessel walls, base and top cap</u>

8.182. Cracks less than 0.5mm in width not associated with changes in section or areas of high stress combination are generally categorised as not significant. Cracks between 0.5mm and 1.0mm in width are monitored for abnormal crack development. Cracks greater than 1.0mm in width are assessed utilising demountable mechanical strain gauges and photography with reference being made to the vessel analysis and/or PPT for evidence of possible high areas of stress.

8.183. At one particular PCPV site, the fixity of the steam penetrations that pass through the side walls of the upper PCPV have been strengthened by the additional of lateral restraint systems (incorporating large diameter under-cut anchors post drilled into the concrete vessel).

8.184. These modifications have been implemented as a preventative measure to mitigate the consequences of any future degradation of the penetration retention welds in the presence of PVCW leaks.

(ii) <u>Mass/reinforced concrete foundation disc / walls and stressing galleries including water stops, joint fillers and sealants</u>

8.185. Leak sealing (via injected grout technique) and refurbishment of water seals in the concrete floor and wall structures of the lower stressing galleries is an option that has been implemented to prevent excessive groundwater ingress.

(iii) Prestressing tendons, ducts and anchorages

8.186. The following activities are undertaken as preventative measures on the pre-stressing system

- New replacement strands are introduced and tensioned to design or pre-approved level (with exception of two of the older PCPVs as discussed earlier).
- Inspected tendon anchorages are routinely coated in fresh grease if required.
- Circumferential wire windings (at podded PCPV sites) are subject to recoating with fresh grease upon their inspection and the associated Change in Load Indicator system that measures the residual horizontal pre-stress (via a series of load cells reacting against the lateral pressure of the windings) is subject to periodic overhaul and replacement.
- Refurbishment and calibration of the tendon stressing equipment between surveillance campaigns.

(iv) Internal insulation

8.187. The PCPV safety case analysis considers the effect of a loss of insulation panel on the concrete structure as one of the infrequent fault load cases. No specific preventative measures exist for the internal thermal insulation. However, an understanding on the aged condition of the insulation is informed by testing undertaken on representative insulation packs exposed to reactor gas and heat in autoclaves, which are periodically tested. Any significant degradation of insulation should be identified early.

(v) Pressure vessel cooling system pipework inside the vessel

8.188. PVCW leaks represent the largest credible risk to the PCPV components and as such the AMP is more developed in this area. For example, AMP strategies for managing PVCW leaks have prioritised the development of a diverse toolkit of techniques for sealing PVCW leaks, in response to known difficulties.

8.189. A high viscosity variant of the usual sealant (Zoric) has recently been developed that has advantages in the reduced number of attempts required to affect a good seal and the potential to seal larger holes.

8.190. Investment has been made in researching sealing techniques that use diverse, mechanical alternatives to the “chemical” based techniques currently in use. This work is still under development and to date the research has produced a lot of learning about the manner in which sealants take effect and the accessibility, or otherwise, of the PVCW network for remote techniques.

8.191. Notwithstanding, the AMP would benefit from utility feedback and experience in the remote sealing techniques of small diameter (inaccessible) cooling water pipework (see paragraph 8.215).

8.192. The forward AMP strategy for PVCW leaks is now primarily focused into making more effective and efficient use of the currently-available sealants. The use of pipe isolations (implemented in the past when prompt leak sealing was considered too challenging) reduces cooling locally to the concrete and without careful control of the environment within the pipe, can exacerbate its degradation. Careful application of the existing leak sealing process has enabled the majority of these isolated pipes to be returned-to-service.

8.193. As part of the drive to improve the efficiency of the PVCW leakage resolution process a sealant that can be cured faster than Zoric (in less than 24 hours) is currently under development.

8.194. Efficiency has also been introduced into the process via the use of digital pressure transducers that reduce the time needed for leak searching tests, from hours down to minutes. This change is being incorporated into the training courses for leak searching.

8.195. Beyond the improvements being developed to improve the physical searching and sealing of pipework, further preventative measures include the application of close chemistry control and temperature monitoring on the cooling water systems used in the PCPVs and flowing through the extensive network of embedded pipework that provides cooling to the liner, standpipes, penetrations and concrete.

8.196. Both the chemistry and temperature are monitored on a regular shift basis to ensure that the PCPVs are operated within a safe range of parameters as mandated by the station operating procedures.

8.197. Measurements of main reactor coolant (CO_2) bulk leakage are regularly made by station staff and the results are periodically reviewed to check that the pressure boundary is intact and that hot gas cannot affect the performance of the main structure.

8.198. The potential for carbon dioxide leakage in the various PCPV designs and the significance of this for vessel integrity has been the subject of specific assessments which were principally based on the effect of any leakage of hot coolant on pre-stressing tendons.

8.199. In some of the later PCPV design layouts additional allowance was made for measurement of CO_2 leakage by the provision of unlined monitoring ducts which pass vertically through the concrete. These ducts are checked regularly as part of the vessel monitoring regime including remote camera inspection to assess any blockages.

(vi) Rubber bearing pads supporting the vessel

8.200. Degradation of the rubber bearing pads is largely prevented by maintaining an acceptable environment at the site of the bearings. The generally passive, confined and benign environment ensures that the rubber material is not exposed to harmful effects that could exacerbate degradation. At some PCPV sites, the bearings are further protected behind steel plating, fire protection or hot gas release insulation.

Remedial Actions

8.201. Remedial measures tend to fall within two categories (i) emergent defects and (ii) safety case improvements.

(i) Prestressed concrete vessel walls, base and top cap

8.202. At the request of the APEX, any significant structural defects identified from the surveillance examination are rectified prior to return to service following the outage (or before the next outage inspection depending on defect significance and access). The defects are typically minor in nature and generally involve local mortar patch repairs around construction joints or adjacent to penetration bearing plates; reinstatement of any missing relief values on tendon cover boxes (where appropriate to PCPV design); refurbishment of top cap survey discs on the upper concrete surfaces adjacent to the standpipe zones (which can become detached due to impact or other activities in the locale).

(ii) Mass/reinforced concrete foundation disc / walls and stressing galleries including water stops, joint fillers and sealants

8.203. Instances of ground water ponding in the lower stressing galleries may be subject to pump extraction to restore notionally dry conditions to this area. For those PCPVs where lower shrouds / oil catchers are fitted, these will be reinstated and/or replaced if loose or damaged.

(iii) Prestressing tendons, ducts and anchorages

8.204. During the tendon surveillance activities the anchor plate bleed holes are rodded to ensure drainage capability and any defective anchorage components are identified for repair / replacement.

(iv) Internal insulation

8.205. No intrusive remedial maintenance is undertaken for the internal thermal insulation. The obsolescence and refurbishment of thermocouples on the liner and penetrations are covered under the wider AMP for Instrumentation and Control.

(v) Pressure vessel cooling system pipework inside the vessel

8.206. The most critical emergent defect on the PCPVs is usually associated with PVCW leaks from embedded cooling pipework. The water leak can pass through the micro-structure of the concrete and reach open voids e.g. tendon ducts, monitoring ducts which can facilitate wider spread of the effects and ultimately contact with steel components e.g. liner, penetration standpipes, pre-stressing tendons, reinforcement.

8.207. The safety case improvements are typically shortfalls in a safety case claim that require to be addressed to maintain adequate design integrity. As an example, significant modifications have taken place on the podded PCPV designs to provide supplementary horizontal pre-stress to the closure concrete units and a dedicated vertical restraint structure has been erected to bolster the vertical resistance to vertical movement of the closure in fault conditions. These extensive modifications were in response to PVCW leaks affecting the steel components of the closure units.

8.208. The unique layout of the concrete closure units on the podded PCPV designs presents some additional risks from PVCW leaks. As a consequence, the fleet-wide risk perspective is largely focused on these particular units. The risks are associated with wetting of the closure holding down bolts; possible degradation of CO_2 pipework that provides pressurising to the interface seals (mating surfaces) between the vessel and closure; degradation of small bore impulse (sampling lines) penetrations embedded in the concrete walls; and a large number of “wetted” tendons that have been adversely exposed to PVCW leaks due to their vertical orientation.

8.209. This has led to a number of initiatives that are focused on possible repair strategies to mitigate the individual and collective risks on these PCPVs. For the “wetted” tendons, remedial activities has included their complete removal for examination and testing to understand any corrosion mechanisms as a result of PVCW wetting (which tends to flow quickly down the ducts and thus displacing the grease protection layer). Mechanisms identified to date that have subsequently affected the wetted tendon strands include general surface corrosion, localised pitting of the tendon strand, and evidence of stress corrosion cracking in the presence of emulsified grease. In general, for small leaks the grease protection continues to provide adequate corrosion prevention. The podded PCPVs provide the bounding case for the number of tendons that have been removed and replaced due to PVCW wetting.

8.210. The podded PCPV risks are effectively safety case shortfalls that were originally identified from surveillance activities to monitor the vessel condition. The issues are attributable to PVCW leakages that had passed through the concrete and contacted critical steel components or the pre-stressing system. The enhanced AMP for the podded PCPVs now in place reflects the greater risks arising from this effect.

(vi) Rubber bearing pads supporting the vessel

8.211. No significant degradation of concern to rubber bearing pads has been encountered to date and hence no remedial actions have been required.

8.2 Licensees' experience of the application of AMPs for PCPVs

8.212. The text in this section of the report has been prepared by the EDF-NG, with only minor changes as described in paragraph 1.9 to 1.10.

8.213. The comprehensive surveillance measures in place are intended to identify any potentially adverse ageing and degradation processes in a timely manner for correction before they present a threat to the long term integrity of the PCPV structures.

8.214. The EDF-NG AMP approach relies upon a series of integrated processes that primarily rely upon robust inspection and surveillance programmes. This approach confirms the material condition and integrity of the PCPV to continue its safety function. Adverse findings that are out with of the predicted expectations or visual evidence that is obtained for deteriorating material condition are either addressed by routine maintenance or specific improvement measures.

8.215. To date the ageing phenomena and long term structural behaviour of the PCPV has aligned with the design expectations e.g. pre-stressing loads, concrete condition, structural performance during operation, grease protection, structural stability of foundations and resilience of the structure to general ageing mechanisms.

8.216. In general the features recorded to date do not represent significant degradation of the condition of the structure and do not raise any concerns for its continuing integrity and fitness for service.

8.217. The type and number of indications and defects observed are also noted to be similar to those observed during previous inspections. This indicates that no significant step changes or increasing degradation in the condition of the concrete structures is taking place.

8.218. The surveillance strategies that underwrite the AMP have gradually changed in recognition of the changing risk profile for the PCPVs as they enter their later generation lifecycle and also based upon operational experience of the structures over many years of investigating issues arising from the in-service inspections:

- Enhanced focus on PVCW leak sealing techniques (including remote access).
- Development of improved PVCW pipe defect repair materials (faster cure time/higher viscosity).
- Improved record and trending of PVCW leaks via a fleet-wide database.
- Improved understanding of tendon duct losses due to friction effects.
- Improved understanding of the resilience of tendon grease to ageing.
- Revised criteria for PCPV tilt limits to reflect nuclear safety risk.

- Deployment of remote video probes on vertical tendons to inspect the in-situ condition.
- Greater recognition that embedded instrumentation provides value to lifetime assessment.
- Enhanced focus of the structural duty and safety case claims on rubber bearing pads and forward inspection requirements.
- Development of a lifetime strategy to prioritise company effort on the PCPV risks pertinent to ageing.

Recommendations by EDF-NG

8.219. It is judged that the current strategy and measures in place provide an adequate AMP for the PCPV structures. EDF-NG continues to seek an improved understanding of the ageing mechanisms and this review has identified some common industry areas where wider information sharing will provide benefit to the longer term ageing management strategy for the PCPV structures. It is therefore proposed that the following topics are raised for discussion during the 2018 topical peer review workshops:

- Feedback and experience on long term rubber bearing pad performance beneath nuclear structures.
- Utility experience and feedback on the effects of long term irradiation exposure of structure concrete to underwrite the current understanding and assessment of lifetime risk to the PCPV integrity.
- Feedback and experience in the application of continuous monitoring of concrete surface cracks using remote electronic techniques that could improve upon the use of discrete point-in-time demountable mechanical strain gauge readings.
- Feedback and experience in the remote sealing techniques of small diameter (inaccessible) cooling water pipework embedded in structural concrete.

8.3 Regulator's assessment and conclusions on ageing management of PCPVs

8.3.1 Criteria and standards for assessing ageing management of PCPVs

8.220. The relevant criteria and standards adopted within this assessment are principally ONR's Safety Assessment Principles (SAPs) and Technical Assessment Guides (TAGs), together with relevant national and international standards and relevant good practice informed from existing practices adopted on UK nuclear licensed sites. In addition to the general criteria and standards described in Section 2 of this report, ONR has utilised the following in the assessment of ageing management of PCPVs:

- ONR Safety Assessment Principles on civil engineering (Ref 3)
- ONR Technical Assessment Guides:
    - Civil Engineering (Ref 52)
    - Civil Engineering containments for reactor plant (Ref 53)
- International Atomic Energy Agency:

- Safety Report Series 82 - Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (Ref 54)
- Nuclear Energy Series NP-T-3.5 - Ageing Management of Concrete Structures in Nuclear Power Plants (Ref 55)

- British Standards Institution:
  - BS 4975 Specification for pre-stressed concrete pressure vessels for nuclear engineering (Ref 60)
  - BS EN 1337-10 – Structural Bearings – Inspection and Maintenance (Ref 61)
- Electric Power Research Institute (EPRI):
  - Program on Technology Innovation: Concrete Civil Infrastructure in United States Commercial Nuclear Power Plants (Ref 59)

8.3.2 ONR's assessment of EDF-NG's ageing management of PCPVs

8.221. EDF-NG's methods for selecting components within the scope of its AMP are defined in the relevant safety cases, which identify the structures, systems and components (SSC) important to safety and the associated claims on the integrity of the PCPVs. Based on the safety claims, the safety cases identify requirements for examination, inspection, maintenance and testing (EIMT). ONR regularly inspects the adequacy of EDF-NG's arrangements for EIMT of the PCPVs and considers that the methods used to select SSCs for inclusion in its AMP are adequate.

8.222. ONR has considered relevant guidance with respect to ageing management programmes (Ref 54 and 55) and has identified one omission from the licensee's AMP. This omission, which was also identified in Chapter 7 in respect of the Sizewell B PWR, relates to the embedded strain gauges used to monitor strains and temperatures in the PCPVs. Although the gauges are not claimed in the safety cases, they provide information of relevance to ageing management of the PCPVs and ONR consider this omission to be an area for improvement as described in Section 8.3.3.

8.223. The main procedures used by EDF-NG to identify ageing mechanisms are station-specific instructions, referenced in the safety case. The licensee's instructions are based on the requirements of BS 4975 (Ref 60), which ONR considers to be an appropriate technical standard for PCPVs. In order to provide further supporting information, EDF-NG has supplemented the minimum surveillance requirements of BS 4975 with other appropriate activities. ONR has seen evidence from its inspections that the procedures are regularly reviewed in response to both operating experience (including from other stations in the fleet) and observed ageing effects.

8.224. An Appointed Examiner (APEX), who is independent of station operations and the safety case justification, manages the civil engineering surveillance programme and reports on their findings. ONR considers that the use of an independent suitably qualified and experienced APEX in this role is in accordance with relevant good practice. As part of its permissioning regime under the licence conditions, ONR regularly assesses the adequacy of the APEX's examination reports for the concrete structures. Based on these assessments, ONR is satisfied that the APEX's surveillance programme identifies ageing mechanisms in the concrete structures and that EdF-NG's process for appointing the APEX results in suitably qualified and experienced persons being appointed.

8.225. EDF-NG invokes its Safety Case Anomalies Process (SCAP) where an examination, inspection, maintenance or testing activity reveals a plant condition that challenges the ageing and degradation assumptions within its safety case. Where these challenges are significant, the licensee may need to make modifications to the safety case, using the design change process to justify continued operations. For the concrete structures, ONR has experience of assessing the safety justifications produced by EDF-NG when changes to expected conditions have been identified (for example in relation to the level of corrosion in tendons). ONR's experience is that the SCAP and design change processes are adequate for capturing ageing-related changes in the concrete structures that are outside the safety case parameters.

8.226. EDF-NG has developed a TLMS to support effective ageing management of the PCPVs beyond their original design life of 30 years. Based on sampling carried out as part of PSR assessments, ONR considers that the PCPV TLMS identifies all significant ageing related risks to be expected at end of life and is adequate.

8.227. EDF-NG has described the outputs from its ageing assessments, including the identification of ageing mechanisms, their significance and the applicable acceptance criteria. The conclusions of ONR's assessment of the evidence relating to EDF-NG's ageing assessment of the PCPVs are given below for each case identified for inclusion in the NARs.

8.228. Acceptance criteria for concrete degradation mechanisms are generally based on the APEX's judgement, supplemented by specialist advice for defects that are more significant. ONR considers the approach taken to be acceptable. The effects of irradiation on concrete are an area of active international research. EDF-NG relies here on guidance in BS 4975: 1990. ONR considers that a review of more recent work on this topic would be beneficial to the AMP and consider this to be an area for improvement as described in Section 8.3.3.

8.229. ONR considers that EDF-NG has identified all the expected significant ageing mechanisms affecting the pre-stressing system, although it has not cited its sources. ONR's experience is that EDF-NG uses ASTM standards as the basis for its testing and acceptance criteria for pre-stressing protective grease coating. ONR considers that such an approach is acceptable as it is based on recognised standards.

8.230. EDF-NG has identified a number of degradation mechanisms for the insulation but does not identify the source of its information. Acceptance criteria for the insulation relate to its ability to perform its safety function, i.e. to maintain the liner temperature within the limits defined in the safety case, as measured by thermocouples. ONR accepts this approach due to the difficulties in inspecting the insulation inside the PCPV and the lack of appropriate standards.

8.231. EDF-NG has identified a number of degradation mechanisms for the PVCSs, but does not identify the source of its information. ONR's experience is that the degradation mechanisms are well understood within the EDF-NG APEX community and have been informed by operating experience. One acceptance criterion for the PVCSs is that of no leaks that could damage other components, such as a pre-stressing strand. Additional criteria in connection with the ability of the systems to cool the vessel also apply, such as restrictions on the degree of isolation. These requirements have not been stated, but are included in the safety cases. ONR considers that these other requirements should be identified more clearly in the AMP, and this area for improvement should be addressed as described in Section 8.3.3.

8.232. EDF-NG has not identified any specific ageing mechanisms for the embedded strain gauges, which are used to monitor concrete temperatures and strains. As noted in paragraph 8.222 these gauges have been omitted from the AMP. Whilst acknowledging that some of the problems with the gauges are due to obsolescence, ONR considers that when an AMP for the gauges is developed it would be strengthened by input from instrumentation and control engineers, in order for EDF-NG to improve its understanding of all the relevant ageing mechanisms. ONR considers that this topic should be addressed by the areas for improvement described in Section 8.3.3. In addition we note that no specific acceptance criteria are stated for availability of the gauges and none is claimed in the safety cases. The APEXs review and comment on the availability of gauges as part of their surveillance programmes. ONR accepts this approach, as many of the gauges cannot be repaired.

8.233. With respect to rubber bearings, EDF-NG has not fully described the applicable ageing mechanisms or acceptance criteria for ageing-related degradation. ONR considers that this is an area for improvement, as described in Section 8.3.3. Inspections carried out by EDF-NG at two of its sites used the categorisation system for defects given by BS EN 1337-10 (Ref 61), which ONR considers good practice.

8.234. The APEXs report the results of the PCPV surveillances every three years, to align with periodic shutdowns. ONR has assessed APEX reports for each shutdown as part of its permissioning process under the licence conditions. ONR considers that the surveillance intervals follow the recommendations of appropriate technical standards and guidance, reflect operating experience and are adequate.

8.235. ONR's assessment of the licensee's surveillances has concluded that:

a) Limited concrete cracking and surface spalling on the PCPV walls has been observed. Based on our site inspections, ONR concurs however with the licensee that this degradation is not structurally significant. Certain wider cracks are periodically monitored and the results trended. ONR agrees with EDF-NG that benefit would be gained by obtaining feedback on the operation of continuous monitoring of surface cracks using remote electronic techniques.

b) Ongoing settlement and tilt effects are largely confined to one station. Detailed surveys and trending are carried out at regular intervals and acceptance criteria are clearly defined in the safety cases. ONR considers that the settlement and tilt that have occurred are within acceptable limits, are appropriately monitored and trended, and do not compromise the safe operation of the reactors.

c) ONR considers that EDF-NG carries out checks on the pre-stressing tendon anchorage loads at appropriate intervals and using an appropriate sampling regime. Effective anchorage loads have, as predicted, reduced with time due to ageing effects. Trending of the results has shown that for all reactors, the estimated mean anchorage load has remained above the minimum design load. On one PCPV, it is possible that the estimated anchorage load will fall below the minimum design load prior to the end of generation. ONR considers that a range of options is available to mitigate this potential issue and that EDF-NG's current surveillances are adequate.

d) ONR's experience from its inspections and assessments is that adequate grease protection is important in order to prevent degradation of the pre-stressing strands. EDF-NG has an adequate procedure for grease testing that aligns with industry good practice. However, except for two stations, grease testing is not a safety case requirement. ONR considers that formalising grease testing as part of the AMP for all EDF-NG's AGR reactors would be worthwhile and so this is identified as an area for improvement in Section 8.3.3.

e) EDF-NG adopts enhanced surveillance for vessels with a history of cooling water leakage, as this has led to strand degradation. EDF-NG's inspections (using remote video probe through the breather hole in the anchorage plate) have provided useful information on strand and grease wetting and degradation and have allowed for the development of a risk-informed approach to strand testing and replacement. ONR considers the approach taken to be in accordance with relevant good practice; in particular with the general approach to ageing management in the IAEA guidance in NS-G-2.12 (Ref 17).

f) ONR considers that PVCS leaks are one of the main degradation mechanisms affecting the PCPVs, and supports EDF-NG's preferred position, which is to only operate the reactors where there are no known leaks and only allow a return to service after a shutdown when all leaks discovered have been adequately sealed. ONR considers that EDF-NG has carried out extensive research and trials to develop and hone its leak searching and sealing techniques and that the extent of leakage on the reactors has significantly reduced because of this work.

g) The number of operational strain gauges continues to fall, and coverage on some reactors is significantly less than on others. ONR accepts that the gauges were not designed for long-term monitoring, however they have provided useful information, including identifying elevated concrete surface temperatures at one station. EDF-NG has recently trialled portable logging equipment to obtain readings from gauges previously considered non-operational. ONR considers that the benefits from this work should be extended, as appropriate, to other stations and so is included as an area for improvement in Section 8.3.3.

h) The bearings supporting the PCPVs are not routinely subjected to visual examination and measurement. Although not intended primarily for nuclear applications, ONR considers that BS EN 1337-10 (Ref 61) contains appropriate guidance in relation to surveillance requirements for bearings and that the standard emphasises the importance of a bearing inspection programme. Whilst acknowledging there will be practical difficulties, ONR considers that periodic visual inspections of bearings should be included in the AMP and that this should be addressed as an area for improvement as described in Section 8.3.3.

8.236. ONR's experience from its inspections and assessments is that the licensee carries out a range of preventive and remedial actions for the PCPVs. For example, a key preventive action is the timely identification and rectification of PVCS leaks. Where EDF-NG identifies defects, they are either routinely addressed during shutdowns (for example replacing pre-stressing tendons) or are raised as APEX recommendations.

8.237. In terms of criteria for taking action when defects are identified, these range from the prescriptive requirements of the safety case (e.g. some aspects of addressing PVCS leaks such as tolerability of leaks) to the use of judgement by the APEX when deciding on the replacement of pre-stressing strands that have been subject to wetting. In taking remedial action, EDF-NG follows its modifications process, the implementation of which is regularly assessed by ONR.

8.238. ONR's experience is that the APEXs raise appropriate recommendations but these are not always implemented within the timescales envisaged by the APEX due, for example, to the constraints of shutdown programmes. In such cases, ONR is nevertheless satisfied that EDF-NG adopts an appropriate risk-informed approach to prioritising defect rectification.

8.239. ONR considers that corrosion found in the Boiler Closure Unit (BCU) pre-stressing wires at Hartlepool and Heysham 1 is important OPEX not described by the licensee in its NAR sections. The corrosion, due to undetected cooling water leakage, led to design and inspection improvements which included the provision of an inert environment around the pre-stressing wires, a leak detection and water extraction system and a pre-stress lock-in system to maintain pre-stress in the event of wire breakage. ONR considers that learning from these problems led to enhanced licensee procedures for pre-stressing inspections and for the detection and sealing of cooling water leakages in difficult-to-access areas.

8.240. Overall, ONR considers that the actions taken by EDF-NG in response to the identification of defects have been duly informed by its safety cases and operating experience and have been adequate.

8.3.3 Overall conclusions on ageing management of concrete pressure vessels (AGR)

8.241. Based on its assessment, ONR considers that EDF-NG's SSC-specific AMP is adequate.

8.242. EDF-NG has raised four recommendations identifying where it considers that wider information-sharing could provide benefit to its longer term ageing management strategy and which should be discussed during the Topical Peer Review during 2018. ONR supports EDF-NG's recommendations and considers that any learning from the Review should be included in revised EDF-NG company guidance as identified in the appropriate areas for improvement. The areas for wider information sharing identified by EDF-NG are:

- Long-term rubber bearing pad performance beneath nuclear structures;
- Effects of long-term irradiation exposure of concrete;
- Continuous monitoring of concrete surface cracks using remote electronic techniques; and
- Remote sealing techniques of small diameter (inaccessible) cooling water pipework embedded in structural concrete.

8.243. ONR's assessment has identified a number of strengths in EDF-NG's AMP, in particular the importance it gives to the role of the independent APEX and its approaches to the detection, categorisation, searching and sealing of vessel PVCS leaks.

8.244. In spite of having an adequate AMP, this assessment has identified a number of areas where improvements to ageing management would be worthwhile. For pre-stressed concrete pressure vessels, these are as follows:

- EDF-NG should provide fleet-wide guidance on the ageing management of strain gauges by the end of 2018 (paragraph 8.222 and 8.232 and 8.235g)).
- EDF-NG should review and make any necessary improvements to its guidance relating to ageing mechanisms and acceptance criteria for PCPV SSCs by the end of 2018 to ensure that it is comprehensive and based on relevant good practice (paragraph 8.231).
- EDF-NG should review and make any necessary improvements to its guidance in relation to the prediction, detection and mitigation of ageing

effects due to the irradiation of concrete by the end of 2018 (paragraph 8.228).

- EDF-NG should review and make any necessary improvements to its fleet-wide arrangements in relation to providing additional evidence, based on an appropriate visual inspection regime that rubber bearings have not been subject to significant ageing effects. The review should be completed by the end of 2018 (paragraph 8.233 and 8.235h)).

- EDF-NG should formalise requirements for regular fleet-wide tendon grease testing by the end of 2018 (para 8.235d)).

8.245. To ensure that these improvements are implemented in a timely manner they have been brought together along with those from other chapters in Chapter 9. Chapter 9 identifies a single area for improvement (AFI) for each licensee to undertake a programme of improvement and the individual elements necessary to complete that programme are clearly identified, along with dates for their completion.

8.246. The overall conclusion from our assessment of PCPVs is that EDF-NG has adequate AMPs, but that the licensee needs to make a limited number of secondary beneficial improvements.

9 Overall assessment and general conclusions

9.1. There are two licensees for the UK fleet of nuclear installations within the scope of this report:

- EDF-NG is the licensee for 15 operating reactors on 8 stations. These have Started operation between 1976 and 1995.
- NNB is the licensee for the twin reactor EPR^{TM} station currently under construction at Hinkley Point C.

The licensees therefore have reactors that are at very different stages of their lives and the provisions for ageing management are consequently also very different.

9.1 Operating reactors

9.2. The EDF-NG reactors have been operating for a significant length of time and are due, at the time of writing, to start permanent shutdown from 2023 onwards. The company therefore has a mature ageing management programme. The AMP is not a standalone process, but integrated into the company's management system. This is consistent with UK regulatory expectations and requirements.

9.3. The elements of the EDF-NG AMP have been presented in a form which demonstrates consistency with the approach described in the IAEA guidance in NS-G-1.12.

9.4. In its assessment of the overall AMP, ONR has concluded that EDF-NG has an adequate overall AMP. Nonetheless the assessment in this report has identified that there are a number of secondary areas where improvement to the overall AMP would be beneficial and proportionate.

9.5. In addition, the assessment of the cases identified for inclusion in this NAR, which considers the application of EDF-NG's AMP to specific SSCs within Chapters 3 to 8 of this report, has demonstrated to our satisfaction that the AMPs for these SSCs are also adequate, although again areas where the licensee can make beneficial and proportionate improvements have been identified.

9.6. None of these improvements alone, nor if considered together, indicate a significant shortfall in EDF-NG's process for ageing management. They have therefore been grouped together into a single area for improvement for the operating reactors, which is :

- AFI-EDF-NG – EDF-NG should improve its ageing management programmes for operating reactors by 30 June 2020, by completing the following:

No	Improvement	Completion date
Overall AMP		
EDF-NG 1	Issue a formal company guidance document to describe the ageing management arrangements, described in Chapter 2 of this report	31 March 2018

No	Improvement	Completion date
EDF-NG 2	Incorporate a review against relevant IGALL documents in the periodic review and update of documents in the technical governance process. This will be in two steps: • Update the technical governance document review management process by 28 February 2018; • First review and update of all technical governance documents by mid-2021.	30 June 2021
EDF-NG 3	Review the arrangements for the annual reporting of ageing management, and include ageing management within an appropriate oversight process for each station for its reviews of performance for the calendar year 2018.	31 December 2018
Electrical cabling		
EDF-NG 4	Review implementation of the technical guidance note for cable condition monitoring at the operating reactor sites to identify areas of good practice and make any necessary improvements to ensure consistent implementation.	31 December 2018
EDF-NG 5	Review the advice given within the technical guidance note for cable condition monitoring to ensure that appropriate advice is given for Neutron Flux Detection cables and make any necessary improvements.	31 March 2018
EDF-NG 6	Update the existing Cable Conditioning Monitoring Technical Guidance Note to differentiate between power cables and I&C cables.	31 March 2018
Concealed pipework		
EDF-NG 7	Review the existing buried pipework inspection strategy and update governance and technical guidance documentation to align and consolidate it within the existing fleet-wide Corrosion Management Programme.	31 December 2018
Reactor pressure vessels		
	Nothing specific identified	
Concrete containment structures		
EDF-NG 8	Provide guidance on the ageing management of PCCV strain gauges.	31 December 2018

No	Improvement	Completion date
EDF-NG 9	Review and make any necessary improvements to the company guidance relating to ageing mechanisms and acceptance criteria for concrete structures SSCs to ensure that it is comprehensive and based on relevant good practice.	31 December 2018
Pre-stressed concrete pressure vessels		
EDF-NG 10	Provide fleet-wide guidance on the ageing management of PCPV strain gauges.	31 December 2018
EDF-NG 11	Review and make any necessary improvements to the guidance relating to ageing mechanisms and acceptance criteria for PCPV SSCs to ensure that it is comprehensive and based on relevant good practice.	31 December 2018
EDF-NG 12	Review and make any necessary improvements to the guidance in relation to the prediction, detection and mitigation of ageing effects due to the irradiation of concrete.	31 December 2018
EDF-NG 13	Review and make any necessary improvements to the fleet-wide arrangements in relation to providing additional evidence, based on an appropriate visual inspection regime that rubber bearings have not been subject to significant ageing effects.	31 December 2018
EDF-NG 14	Formalise requirements for regular fleet-wide tendon grease testing.	31 December 2018

9.7. As noted above, the nature of these identified improvements does not indicate that there is any generic problem with EDF-NG's approach to ageing management programmes. Under its management system the licensee keeps all elements of its ageing management system under continuous review and makes improvements promptly when they are identified. ONR does not therefore consider that a broader review of EDF-NG's operating reactor Ageing Management Programme is warranted, over and above the reviews undertaken in the present review exercise.

9.2 Hinkley Point C

9.8. Hinkley Point received ONR's permission to start construction earlier this year and is in the design stage of its lifecycle. NNB's AMP is therefore much less developed than for EDF-NG's operating reactors. Based on the IAEA five step model for ageing management in NS-G-1.12, it has only developed the Understand and Plan steps. ONR considers this to be appropriate for Hinkley Point C given the project's current stage of construction and design development.

9.9. Again, similar to the operating reactors, ONR has judged that for Hinkley Point C both the overall AMP and the AMPs for the SSC cases identified for inclusion in this NAR are adequate, but that there are secondary areas where, at this stage in the project, the licensee could make beneficial and proportionate changes to its AMPs. These have been grouped into a single area for improvement for Hinkley Point C, which is:

- AFI-NNB – NNB should improve its ageing management programmes for Hinkley Point C by 31 December 2018, by completing the following:

No	Improvement	Completion date
Overall AMP		
NNB 1	Formalise the ageing management arrangements by producing a Corrosion and Ageing Management Strategy.	30 June 2018
Electrical cabling		
	Nothing specific identified.	-
Concealed pipework		
	Nothing specific identified	
Reactor pressure vessels		
	Nothing specific identified	
Concrete containment structures		
NNB 2	Formalise the ageing management arrangements for the concrete structures, for all stages of construction. The arrangements should describe the methods and criteria used for selecting SSCs to be included in the programme and the processes/procedures for the identification of relevant ageing mechanisms.	31 December 2018
NNB 3	Review and make any necessary changes to ageing mechanisms and acceptance criteria for the liner coating and penetration seals to ensure that these are comprehensive and provide appropriate guidance to enable the development of the material specifications and EIMT activities.	31 December 2018

9.10. As noted above the Hinkley Point C AMPs are at early stages, reflecting the plant still being designed. The AMPs will develop with time under the licensee's management system as the design progresses. ONR's oversight of this will be part of normal regulatory business and hence ONR does not consider that a broader review of NNB's Ageing Management Programme is warranted, over and above the reviews undertaken in the present review exercise.

9.3 Overall conclusions

9.11. ONR's assessment has found that both the UK's operating reactors and its reactors under construction have adequate ageing management programmes appropriate to the stages that they are at in their lifecycles. Notwithstanding this, a number of secondary but beneficial improvements have been identified for both licensees and programmes for improvement have been developed and agreed.

10 References

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2 WENRA Safety Reference Levels for Existing Reactors September 2014
www.wenra.org/media/filer_public/2014/09/19/wenra_safety_reference_level_for_existing_reactors_september_2014.pdf

3 Safety Assessment Principles for Nuclear Facilities 2014 Edition Revision 0
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4 Office for Nuclear Regulation - Permissioning inspection - Technical assessment guides
www.onr.org.uk/operational/tech_asst_guides/index.htm

5 Office for Nuclear Regulation - Compliance inspection - Technical inspection guides
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6 IAEA Safety Standards
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www-ns.iaea.org/projects/igall/

8 Electrical Power Research Institute
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Annex 1 - Abbreviations

Term / Abbreviation	Definition
AFCEN	French Society for Design and Construction Rules for Nuclear Island Components
AFI	Area for Improvement
AGR	Advanced Gas Reactor
AL	Anchorage Load
ALARP	As Low as Reasonably Practicable
AM / AMP	Ageing Management (Programme)
AMS	Asset Management System
ANSI	American National Standards Institute
APEX	Appointed Examiner
ASME	American Society of Mechanical Engineers
ASR	Alkali-Silica Reaction
BAC	Boric Acid Corrosion
BCI	Buried Cast Iron
BMI	Bottom Mounted Instrumentation
BS	British Standard
CANDU	Canada Deuterium Uranium reactor
CMP	Corrosion Monitoring Programme
CCM-TGN	Cable Condition Monitoring Technical Guidance Note
CI	Cast Iron (plant context)
CIV	Containment Isolation Valve
CoF	Consequences of Failure
CLA	Component Life Assessment
Company	EDF Energy Nuclear Generation Ltd
CP	Cathodic Protection
CRDM	Control Rod Drive Mechanism
CTS	Company Technical Standard
CUI	Corrosion Under Insulation
CW	Cooling Water
Dpa	Displacement per atom
EAL	Effective Anchorage Load
EAU	Containment Monitoring System
EDF	Electricite de France
EDF-NG	EDF Energy Nuclear Generation Ltd
EIMT	Examination, Inspection, Maintenance and Testing
ELLDS	End of Life Limiting Defect Size
ENSREG	European Nuclear Safety Regulators Group
EPR	Ethylene Propylene Rubber
EPRI	Electrical Power Research Institute
ER	Equipment Reliability

Term / Abbreviation	Definition
ETA	European Technical Approval
EU	European Union
FA3	Flamanville 3
GDA	Generic Design Assessment
HPC	Hinkley Point C
HDPE	High Density Polyethylene
HSE	Health & Safety Executive
HSWA74	Health and Safety at Work etc. Act 1974
HTTU	Hybrid Trepanning Tool Unit
HV	High Voltage
IAEA	International Atomic Energy Agency
IC	Intelligent Customer
IEC	International Electrotechnical Commission
IGALL	International Generic Ageing Lessons Learned
ILRT	Integrated Leak Rate Test
INES	International Nuclear Event Scale
INPO	Institute of Nuclear Power Operators
IoF	Incredibility of Failure
IR	Insulation Resistance
ISI	In Service Inspection
ISM-CTS	In-Service Management of I&C Cables Central Technical Standard
ISP	Inaccessible Systems Programme (concealed pipework, Chapter 3)
ISP	Irradiation Surveillance Programme (Reactor pressure vessels, Chapter 5)
IVC	Inspection Validation Centre
I&C	Control & Instrumentation
LC	Licence Condition
LFCG	Lifetime Fatigue Crack Growth
LOCA	Loss of Coolant Accident
LoF	Likelihood of Failure
LV	Low Voltage
MDI	Maintain Design Integrity
MDL	Minimum Design Load
MICC	Mineral Insulation Covered Cable
MITS	Maintenance Inspection and Testing Schedule
MS	Maintenance Schedule
MV	Medium Voltage
NAR	National Assessment Report
NDT	Non Destructive Testing
NEI	Nuclear Engineering Institute
NFD-TGN	Neutron Flux Detection Technical Guidance Note
NIA65	Nuclear Installations Act 1965

Term / Abbreviation	Definition
NICIE	New In Core Inspection Equipment
NNB	Nuclear New Build
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission (USA)
NSDAP	(NNB's) Nuclear Safety Design Assessment Principles
NSPs	Nuclear Safety Principles
NSAPs	Nuclear Safety Assessment Principles
OEM	Original Equipment Manufacture
OLP	Organisational Learning Process
OPEX	Operating Experience
ONR	Office for Nuclear Regulation
PCCV	Prestressed Concrete Containment Vessel
PCPV	Prestressed Concrete Pressure Vessel
PD	Partial Discharge
PE	Polyethylene
PEEK	Polyether Ether Ketone
PFA	Pulverised Fuel Ash
PI	Polarisation Index
PLEX	Plant Life Extension
PIE	Postulated Initiating Event
PM	Preventative Maintenance
PPT	Proof Pressure Test
PSR	Periodic Safety Review
PT	Period Testing
PWR	Pressurised Water Reactor
PTFE	Polytetrafluoroethylene
PVC	Polyvinyl Chloride
PVCW	Pressure Vessel Cooling Water
PWR	Pressurised Water Reactor
PWROG	Pressurised Water Reactor Owners Group
QA / QC / QAP	Quality Assurance / Quality Control / Quality Assurance Programme
OPEX	Operating Experience
PCCV	Pre-stressed Concrete Containment Vessel
PCPV	Pre-stressed Concrete Pressure Vessel
PVCS	Pressure Vessel Cooling System
RACW	Reactor Auxiliaries Cooling Water
R&D	Research & Development
RBI	Risk Based Inspection
RCS	Reactor Coolant System
RCW	Reactor Cooling Water
RD	Responsible Designer
RHWG	Reactor Harmonisation Working Group
RPV	Reactor Pressure Vessel

Term / Abbreviation	Definition
SAP	Safety Assessment Principles
SBI	System Based Inspection
SCAP	Safety Case Anomalies Process
SCC	Stress Corrosion Cracking
SFAIRP	So Far As Is Reasonably Practicable
SHIP	System Health Indicator Programme
SIP	Structural Integrity Panel
SOT	Structural Overpressure Test
SQEP	Suitably Qualified & Experienced Personnel
SR	Silicone Rubber
SRCM	Streamlined Reliability Centred Maintenance
SRL	Safety Reference Level
SSC	System, Structure or Component
TAG	Technical Assessment Guide
TEA13	Energy Act 2013
TG	Technical Governance
TGN	Technical Guidance Note
TIG	Technical Inspection Guide
TLMS	Through Life Management Strategy
TLSC	Time Limited Safety Case
TPR	Topical Peer Review
TSA	Thaumasite Sulphate Attack
UT	Ultrasonic / Ultrasonic Testing
WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulators Association
WM	Work Management
XPLE	Cross linked polyethylene