Ministry of Infrastructure and Water Management

Netherlands' National Assessment Report for the Topical Peer Review on Ageing Management

In compliance with EU's Nuclear Safety Directive 2014/87/EURATOM (NSD)

The Hague, December 2017

Executive Summary

This report presents the National Assessment of ageing management programmes of nuclear installations in the Netherlands according to the WENRA TPR Specifications.

The participating nuclear installations are Nuclear Power Plant (NPP) Borssele, the High Flux Reactor Petten (HFR) and the Higher Education Reactor (HOR) of the Delft University of Technology. All installations have an age beyond 40 years and a power above 1MW. The participation of the research reactors is mainly limited to the overall Ageing Management Programme (AMP) and electrical systems and for one of them the concealed piping. The report has been drafted by the Authority for Nuclear Safety and Radiation Protection (ANVS) in close cooperation with the licensees, who provided the input for the chapters dealing with the four Structures Systems and Components - except for the final section of each chapter, presenting the Regulator's assessment and conclusions.

Several elements of what today is called ageing management (AM), used to be part of maintenance, surveillance and periodic testing. The introduction of AM was stimulated by the regulatory body since around 1990, starting with NPP Borssele. In those days the regulatory body in the Netherlands was relatively small and its focus mainly was on the NPPs in operation. The attention for the research reactors and their AM was limited. Around the same time the instrument of Periodic Safety Review (PSR) was introduced also first at the NPPs. It was in the period 1995-2000 that regulatory attention grew for the research reactors, with introducing PSR. AM became a point of attention around 2000 at the HFR and 2010 at the HOR. General licence requirements, IAEA-standards, PSRs and IAEA-missions like OSART, INSARR, AMAT and SALTO were drivers for the gradual development and improvement of AM. Currently the HOR does not (yet) have a licence requirement on AM. However, it will be included, following a licence revision in the next years. The development of AM at the HOR is required in the PSR implementation plan. Both research reactors develop their AM with the IAEA standard SSG-10 as a basis.

At the NPP Borssele in the period 2007-2012 a comprehensive Long Term Operation (LTO) programme was carried out to prepare an application for a modification of the existing licence to accommodate the relevant aspects for the operation beyond the original design life of 40 years till 60 years. Several LTO-related licence requirements are still to be fulfilled till 2020, one of them being the final confirmation of the large margins against embrittlement of the reactor pressure vessel. The AMP of NPP Borssele is mature and complies with the state of the art. One issue of attention is the consequence of the phase-out of nuclear power in Germany. Both ANVS and the licensee already have taken steps and will keep this on the agenda. This is an ongoing issue since several years as already mentioned in reports to the Convention on Nuclear Safety.

At the HFR the AMP was gradually being built up based on an evaluation in 2003-2005 during the 1st PSR. Several AM-related significant events in the period 2008-2013 led to the beginning in 2013 of an ambitious programme of Asset Integrity, Ageing Management Review and since 2016 also LTO was included. A manifestation of this are the increased number of staff in the maintenance department and the recruitment, beginning of 2017, of a new maintenance manager, who is also responsible for AM and has experience in the building of structured programmes. The evaluations and reviews on Asset and Ageing Management have been finished in 2014. Important improvements on the

identified gaps have already been implemented in the last years. In 2017 the last step to further develop the AM into an integrated AMP started, with the goal to be ready for the IAEA SCO (Safety of Continued Operation) mission that has been invited by the ANVS and is to be conducted in 2019 or 2020. To assure this the ANVS will require the licensee to submit a detailed planning with milestones and periodic reporting. After completion the HFR will have a state-of-the-art AMP.

At the HOR, since it is a much smaller and less complicated reactor, the AMP was developed and implemented in a relatively short period (2012-2016), and endorsed as sufficient by ANVS. In 2020 there will be an INSARR mission with special attention on AM in relation with the next PSR.

The coming years ANVS will focus its supervision activities on the remaining LTO issues at the NPP, the implementation of AM at the HFR and the development of a future inspection plan for implemented AMPs.

A number of candidates for good practices are offered:

- The cable depot that was created by the HFR;
- The recruitment at the HFR of a maintenance manager, responsible for the AMP and with experience in setting up of a structured maintenance/ageing management programme according to modern approaches;
- The setup and implementation of the AM at the HOR according to SSG-10 by a small organization in a relatively short time;
- The strong international engagement of the NPP in the area of ageing management;
- The stimulation (together with Belgium) to develop an LTO mission for RRs.

Preamble

This is the National Assessment Report of the Netherlands for the Topical Peer Review 2017 on Ageing Management. The structure and content of this report complies with the requirements as specified in the guideline[1] developed for the Peer Review process led by the European Nuclear Safety Regulators Group (ENSREG). The national report has been prepared by the Authority for Nuclear Safety and Radiation Protection (ANVS[2]) which constitutes the major part of the competent regulatory body in the Netherlands for nuclear reactors.

Background

In 2014, the European Union (EU) Council adopted directive 2014/87/EURATOM amending the 2009 Nuclear Safety Directive to incorporate lessons learned following the accident at the Fukushima Daiichi nuclear power plant in 2011. Recognising the importance of peer review in delivering continuous improvement to nuclear safety, the revised Nuclear Safety Directive introduces a European system of topical peer review which will commence in 2017 and every six years thereafter. The purpose is to provide a mechanism for EU Member States to examine topics of strategic importance to nuclear safety, to exchange experience and to identify opportunities to strengthen nuclear safety. The process will also provide for participation, on a voluntary basis, of States neighbouring the EU with nuclear power programmes.

The 30th Meeting of ENSREG identified *ageing management of nuclear power plants* as the topic for the first Topical Peer Review. This selection was informed by a technical assessment performed by the Western European Nuclear Regulators Association (WENRA) in recognition of the age profile of the European nuclear reactor fleet and the economic and political factors supporting long term operation of European nuclear power plants.

ENSREG in June 2016 decided to *extend the scope of the exercise by including research reactors* with a thermal power larger than 1MW. For the Netherlands this meant the research reactors ‘High Flux Reactor’ (HFR) in Petten and ‘Hoger Onderwijsreactor’ (HOR) in Delft were to be included.

ENSREG coordinates the topical peer review process, supporting cooperation between Member States. WENRA has supported the process by preparing a technical specification[1] to define the expected scope and content of the national assessment reports.

Objectives of the Topical Peer Review process

- Enable participating countries to review their provisions for ageing management of nuclear reactors, to identify good practices and to identify areas for improvement.
- Undertake a European peer review to share operating experience and identify common issues faced by Member States.
- Provide an open and transparent framework for participating countries to develop appropriate follow-up measures to address areas for improvement.

[1] Topical Peer Review 2017 Ageing Management Technical Specification for the National Assessment Report, drafted by the Western European Nuclear Regulators Association (WENRA), Reactor Harmonisation Working Group (RHWG) Report to WENRA, 21 December 2016

[2] Dutch: ‘Autoriteit Nucleaire Veiligheid en Stralingsbescherming’, ANVS

Process Outline; three phases

- National assessment (January – December 2017) - performed by Member States according to a technical specification prepared by WENRA's Reactor Harmonisation Working Group. The present report presents the results of the assessment conducted in the Netherlands.
    - licensees performed a self-assessment in line with the WENRA technical specification;
    - the assessments were independently reviewed by the national regulator, during preparation of the national assessment report;
    - the national assessment report was edited and finalised by the ANVS.
- Peer Review (2018) - including a peer review workshop and publication of a summary report setting out overall findings and ENSREG's proposed follow-up activities.
    - Pre-workshop review of national reports (January – April 2018);
    - Peer Review workshop (May-June 2018);
    - Publication of Workshop Report (August 2018).
- Follow-Up (2018 – 2023) - definition and implementation of measures to address relevant findings from national assessment and peer review process.
    - Publication of ENSREG Implementation Plan (December 2018);
    - Report status of implementation of follow up actions (December 2023).

Scope

As stated above, the scope extends to research reactors with a power larger than 1 MW_{th}.

The assessment process examined the application of the ageing management programmes to the following systems structures and components (SSCs):

- Electrical cables;
- Concealed piping;
- Reactor pressure vessels (or equivalent structures);
- Concrete containment structures.

Contents

Figures

Tables

1 General information

1.1 Nuclear installations identification

The table below contains all reactors participating; one power reactor and two research reactors (RRs). No other reactors fall within the definitions.

Table 1 Nuclear installations identified to be within the scope of the topical peer review

Type of Installation	**NPP**	**RR**	**RR**
Name	NPP Borssele	Hoge Flux Reactor[3] (HFR)	Hoger Onderwijs Reactor[4] (HOR)
Licensee	EPZ NV	NRG	TU Delft
Type of reactor	PWR/2-loop	Tank in pool	Open pool
Power output	1365MWth/515MWe	45MWth	3MWth[5]
Year of first operation	1973	1961	1963
Scheduled shutdown date	2033	n.a.	n.a.

1.2 Process to develop the National Assessment Report

Process

Soon after the decision of ENSREG in June 2016 to include RRs with a thermal power larger than 1MW, the ANVS held a first information meeting with all under 1.1 mentioned licensees. A second information meeting was held in November 2016, where the cooperation of ANVS and the three licensees, including the distribution of the writing and review tasks and the global planning were concluded. Among others, it was decided that certain chapters and paragraphs would be written by the ANVS and others by the licensees. Following that second meeting, in the same month formal letters were sent to the licensees with a request for cooperation according to the agreed conclusions. All licensees sent a positive response.

Within the ANVS a small project team of five people was set up, with experts related to the subjects. Also the three plant inspectors were involved partly. If needed other ANVS people were asked. The project leader (PL) primarily focused at the NPP and the deputy PL focused on both RRs. A steering group of ANVS team coordinators was informed periodically and if necessary was asked to make decisions.

The three licensees created their own project teams.

Most texts created by ANVS were reviewed by the licensees, and vice versa. One central contact point was named per organization. Several meetings were held at contact point level.

[3] High Flux Reactor, operated by licensee NRG in Petten
[4] Higher Education Reactor in Delft operated and owned by licensee Delft University of Technology
[5] Maximum licensed power

Editorial

Since the Netherlands has only one nuclear power plant (NPP Borssele), the subdivision proposed by the guidance provided by WENRA RHWG has been simplified to:

- *NPP Borssele*
- *HFR - specific information*
- *HOR - specific information*

Scope of exercise

The table below shows the scope per chapter of the present NAR.

Table 2 Scope per chapter of the present National Assessment Report

Chapter number	Topic	Applies to facilities
2	Overall ageing management programme requirements & implementation	NPP Borssele, HFR, HOR
3	Electrical Cables	NPP Borssele, HFR, HOR
4	Concealed pipework	NPP Borssele, HFR
5	Reactor pressure vessels*	NPP Borssele
6	Calandria/pressure tubes (CANDU)	*none*
7	Concrete containment structures	NPP Borssele
8	Prestressed concrete pressure vessels (AGR)	*none*

(*) Licensee NRG operating the HFR, volunteered to submit material on their reactor vessel although the HFR does not feature a pressure vessel in the usual sense. Their contribution can be found in Appendix A.

2 Overall management programme requirements and implementation

2.1 National regulatory framework

The national regulatory framework consists of a pyramid containing among others the binding elements Nuclear Energy Act (Kernenergiewet), governmental decrees and ministerial decrees (including those implementing the EU-Nuclear Safety Directive). Furthermore it consists of non-binding elements (guidance documents), e.g. NVRs (adapted IAEA requirements and guides for NPP) and the so-called Dutch Safety Requirements. Lastly there are industrial standards that may be part of the licensing base. Non-binding documents can be made binding through attachment to the licence. Refer to the National Report on the Convention on Nuclear Safety for more detailed information.

NPP Borssele

Since the mid-nineties of the last century a general licence requirement on ageing management is applicable.

The NVRs are applicable for the NPP. The following NVRs are currently relevant for ageing management:

- NVR-NS-R-1 Design
- NVR NS-R-2 Operation
- NVR NS-G-2.6 Maintenance, Surveillance and In-Service Inspection
- NVR NS-G-2.12 Ageing Management for NPPs
- NVR-NS-G-2.10 Periodic Safety Review of NPPs

The WENRA Safety Reference Levels are largely covered in these NVRs.

In addition since 2012 in the licence there is a chapter with a number of licence conditions related to long-term operation (LTO) beyond 2013 (2014-2033). For LTO a modification of the licence was necessary, because the original Final Safety Analyses Report (FSAR) was based on calculations for 40 years. It was agreed between licensee and the regulator that the licensee would carry out an LTO assessment in the period 2008-2012, then followed by a licence application. It would be carried out according to the IAEA Guideline SR57 added by a small number of PSR Safety Factors. The PSR to be delivered end of 2013 was allowed to use the results of these Safety Factor assessments. A few years later a licence revision was made in relation with the implementation of PSR improvements and still applicable LTO provisions were kept in that licence.

Research Reactors

Both research reactors have a general licence requirement for periodic safety review. The HFR also has a licence requirement since 2005 for ageing management (it calls for a roadmap for the development). ANVS has decided that the licences will be upgraded in de next few years to include NVRs for RRs, including one based on IAEA SSG-10 (Ageing Management for Research Reactors). Currently the HFR also has a licence condition to carry out a 5-yearly INSARR mission (including an AM module). Based on the graded approach the HOR licence will also be adapted to include an INSARR mission requirement with a frequency of 10 years, starting in 2020. As part of its LTO-policy the ANVS also agreed with licence holder of the HFR to have an IAEA SCO-mission in 2019/2020 (LTO-mission for RR developed by IAEA in request from Belgium and The Netherlands). Based on the graded approach an SCO mission is not considered necessary for the HOR.

For the implementation of ageing management programmes it has been agreed by the licensees and ANVS to use IAEA SSG 10, the safety guide for the ageing management of research reactors.

2.2 International standards

The Netherlands is a country with a small nuclear programme. Therefore it decided in the past not to develop its own technical regulations, but to make ample use of IAEA-standards (see section 2.1). In particular for ageing management, The Netherlands makes use of the IGALL programme[6]. The licensee of the NPP is member of its steering committee.

It is also important to note that the principle of continuous improvement is embedded in the approaches related to the PSR and the Nuclear Safety Directive.

[6] IAEA Extra-budgetary Programme on International Generic Ageing Lessons Learned (IGALL) for Nuclear Power Plants

NPP Borssele

For the LTO assessment of the NPP the IAEA guideline SR57 has been used. The NPP is actively participating in SALTO peer reviews, workshops and consultancy meetings for creation of guidance in this area.

RR HFR

The HFR ageing management programme (AMP) has been developed over the years 2005-today in several stages and also related to the PSRs. Completion of its Implementation (including LTO and further improving to a state-of-the-art integrated AMP) is aimed in the next years. Currently the SSG-10 is used as a reference. The development is supervised by the ANVS.

RR HOR

The HOR ageing management programme (AMP) has been developed over the years 2012-2016 using the International Atomic Energy Agency (IAEA) Specific Safety Guide No. SSG-10 as a guideline. The development of the AMP is performed under the supervision of the ANVS.

Input and discussions on the way to shape, execute and maintain the ageing management programme were held with some of the HOR peer nuclear research institutes. More specifically the operating teams of the BR2 reactor operated by SCK-CEN and the SAFARI reactor operated by NECSA were consulted.

Comparison of SSG-10 with TPR specification

As the WENRA report “Topical Peer Review 2017; Ageing Management; Technical Specification for the National Asssessment Reports” was provided after the development of the AMP, the ANVS has requested the ‘Gesellschaft für Anlagen- und Reaktorsicherheit’ (GRS) to compare the above mentioned WENRA report to the above mentioned IAEA guide SSG-10. Equivalence between both guides was shown by GRS in the report no. ANVS-WP6-T1, with the subject title ‘Requirements on Ageing Management of Research Reactors’. The differences are limited. Non-significant deviations between the RR AMPs and the requirements in the WENRA report can be attributed to the fact that these AMPs were developed with the IAEA SSG-10 as a guide.

2.3 Description of the overall ageing management programme (AMP)

2.3.1 Scope of the overall AMP

2.3.1.1 Scope of the overall AMP & Ageing management- NPP Borssele

During the design phase already assumptions were made about specific ageing mechanisms like neutron induced embrittlement of the reactor pressure vessel and fatigue of the primary components. Based on that, specific programmes were implemented to monitor these mechanisms.

Like every NPP several generic programmes like maintenance, in-service inspection and surveillance are in place at NPP Borssele which comprise ageing management activities. In the context of overall operational management, activities and responsibilities for these programmes are distributed throughout the organization of the NPP.

The term ageing management was introduced in the nineties particularly in IAEA guidelines based on the fact that most existing NPPs were built 20 to 30 years ago and that new ageing mechanisms have been discovered. It was also realized that effective ageing management of Systems, Structures and Components (SSC) can best be accomplished by coordinating the existing programmes under a systematic 'umbrella' type of ageing management[7].

In the Periodic Safety Review of NPP Borssele performed during 1990 – 1993 specific attention to ageing management was introduced. In the following PSRs comprehensive ageing management reviews have been performed.

In the frame of the PSR of 2000 – 2003 an IAEA AMAT (Ageing Management Assessment Team) Peer Review was performed to look for further improvement on ageing management.

As part of the regulatory process for Long Term Operation of NPP Borssele IAEA SALTO Peer Reviews have been performed in 2009 and 2012 and a SALTO follow-up Peer Review in 2014.

Ageing Management Team and Ageing Experience Feedback Procedure at NPP Borssele

Based on the PSR performed in 1990 – 1993, in the nineties a requirement to have an adequate ageing management system was included into the licence of NPP Borssele. To fulfil this requirement a specific procedure, PU-N12-19[8] was introduced in the management system to assess ageing related in- and external events. In parallel with the implementation of this procedure an Ageing Management Team (AMT) was introduced responsible for the ageing assessments.

The AMT is part of the technical support department (KTE). The task of the AMT is to gather relevant information related to SSC ageing (from both internal and external experiences) and to keep up to date with new insight and knowledge in relevant technical areas. External sources are for example WANO SOERs and SERs, IRS event reports, VGB event reports and generic assessments of the OEM. This process is schematically illustrated in Figure 1. The purpose is to ensure that all newly received information is acknowledged and assessed accordingly and that appropriate actions are undertaken. This process is supported by a specific ageing management database in which all information is registered. Based on this procedure several actions have been taken including enhancing of existing programmes.

[7] Implementation and Review of a Nuclear Power Plant Ageing Management Programme, Safety Report Series No. 15, 1999

[8] [8] 'Het analyseren en evalueren van verouderingsmeldingen en van wijzigingsmeldingen', EPZ Uitvoeringsprocedure PU-N12-19, 2-3-2016

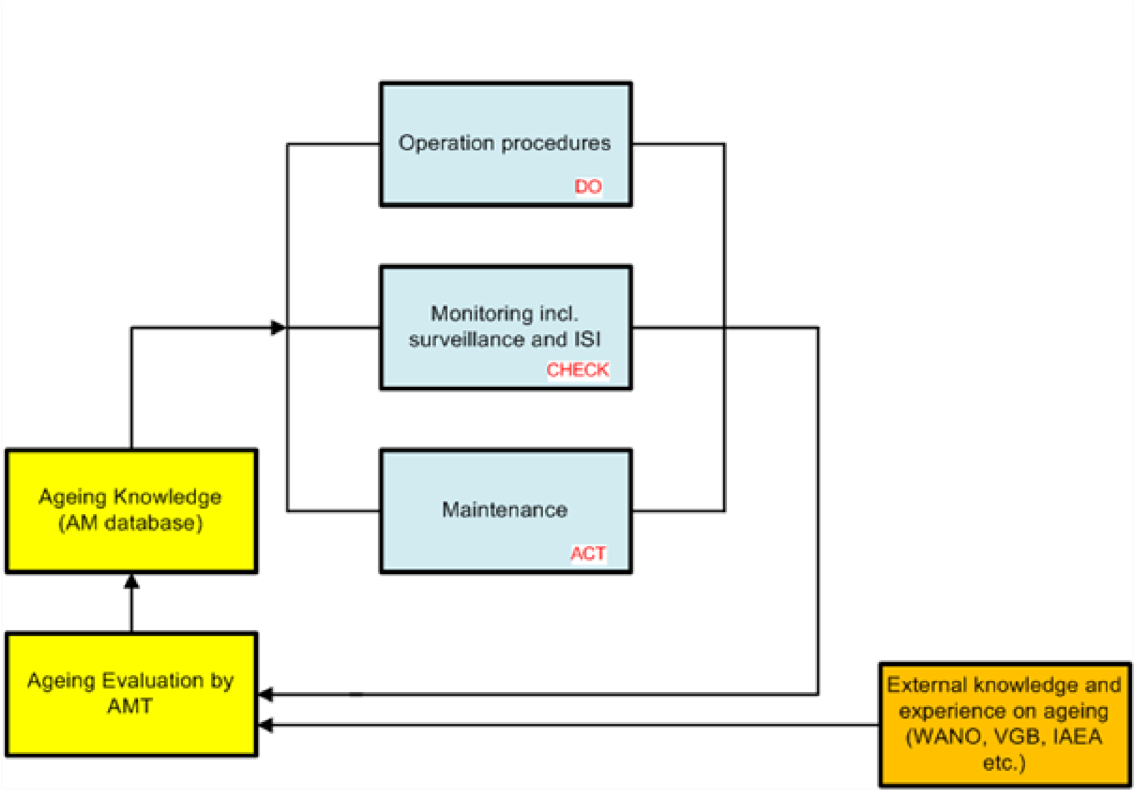


Figure 1 Overview of ageing management feedback procedure at Borssele NPP (PU-N12-19[9]) at NPP Borssele

The AMT is also responsible for establishing generic Ageing Management Reviews (AMRs). Two comprehensive AMRs have been carried in the framework of the Periodic Safety Review (2000-2003) and the Long Term Operation Assessment (2010-2012[10]). In both AMRs passive structures and components important to safety have been reviewed.

As will be mentioned further, the AMR performed for the LTO Assessment is used as a basis for the current overall Ageing Management Programme and therefore first the AMR is dealt with below.

Ageing Management Review at NPP Borssele

Scope

The scope was determined based on scoping and screening process based on IAEA safety report 57[11].

According to IAEA SR57 the Systems Structures and Components (SSCs) within the scope of LTO assessment are the following:

1. All SSCs important to safety :
    a. That ensure the integrity of the reactor coolant pressure boundary;
    b. That ensure the capability to shut down the reactor and maintain it in a safe shutdown condition;

[9] Het analyseren en evalueren van verouderingsmeldingen en van wijzigingsmeldingen, EPZ Uitvoeringsprocedure PU-N12-19, 2-3-2016
[10] Conceptual document LTO “Bewijsvoering” KCB, NRG-report NRG-22701/10.103460, 9 September 2011
[11] Safe Long Term Operation of Nuclear Power Plants, Safety Reports Series No. 57, 2008

   c. That ensure the capability to prevent accidents that could result in potential off-site exposure or that mitigate the consequences of such accidents.

2. Other SSCs whose failure may impact upon the safety functions specified above.

Based on these criteria, the SSCs in the LTO scope were identified and reported in close cooperation between EPZ and AREVA (successor of Siemens/KWU, the designer of NPP Borssele). The safety functions of these SSCs were identified in detail, and subsequently categorized in three "safety categories":

- Safety category 1, This category contains components of the reactor coolant system whose postulated catastrophic failure is not enveloped by accident analyses. In the event of postulated catastrophic failure (for example circumferential break at a weld) of the main components of the reactor coolant pressure boundary an event sequence is to be expected for which accident control has not been verified. For this reason, these components are assigned to category S1.
- Safety category 2, Other SSCs important to safety, including
  - high-energy SSCs inside the containment whose postulated failure may lead to cross-redundancy consequential damage, or
  - whose failure initiates a design-basis accident with immediate adverse impact on heat removal from the reactor core;
  - SSCs for the control of design-basis accidents (safety functions), for which no alternative measures are available promptly or in an adequate time frame;
  - SSCs with auxiliary/supply functions whose failure will lead to loss of safety functions required for accident control;
  - Supports as well as supporting structures for category 1 components.
- Safety category 3, SSCs, whose failure may impact upon the safety functions specified in categories 1 and 2.

The scoping results are limited to system level.

Screening process

In the screening phase further detailing of the different SSCs on structure and component (SC) level is performed and active and passive SCs are identified. The passive SCs identified in the screening step are subject to AMR. The active SCs are subject to a review of existing plant programmes in the active components assessment[12].

- Passive SCs are structures, components or subcomponents whose functioning does not depend on an external input such as actuation, mechanical movement or supply of power;
- Active SCs are defined as structures, components or subcomponents that are not passive.

The passive SCs resulting from screening for NPP Borssele are classified into four main groups in order to facilitate the AMR. These four groups are:

[12] Conceptual document LTO "Bewijsvoering" KCB, NRG-report NRG-22701/10.103460, 9 September 2011

- Mechanical A SCs are defined as forming part of the fission product barrier, i.e. the barrier preventing radioactive release to the environment:
    - Reactor Pressure Vessel;
    - Pressurizer;
    - Steam Generators;
    - Main Coolant Pumps;
    - Control Rod Drive Mechanism pressure housings;
    - Main Coolant Lines and pressurizer surge line;
    - Steel containment.
- Mechanical B SCs consist of the remaining in-scope mechanical systems:
    - Nuclear safety systems;
    - Safety-related auxiliary systems;
    - Secondary systems;
    - Heating, Ventilation and Air-Conditioning systems;
    - RPV internals;
    - Primary component supports;
    - Remaining in-scope supports and hangers;
    - Mechanical fasteners.
- Electrical commodity groups:
    - Cables;
    - Wires;
    - Electrical connectors;
    - Electrical Containment Cable Penetrations;
    - Cable penetrations (floors, walls);
    - Cabinets, racks;
    - Cable trays.
- Structural commodity groups:
    - Reactor Building;
    - Auxiliary Reactor Building;
    - Secondary Pressure Relief Station;
    - Turbine Building;

- Switchgear Building;
- Ventilation Stack;
- Emergency Diesel Generator Buildings;
- Building for Auxiliary Water Supply Systems;
- Emergency Control Building;
- Cooling Water Inlet Building;
- Cooling Water Outlet Building;
- Pipe and Cable Ducts;
- Cranes and Lifting Equipment;
- Fire Protection Barriers.

For these passive SC a comprehensive AMR was performed. For the active SC resulting from scoping & screening a review was performed to check and show if the safety functions of these SC are adequately incorporated in a preventive maintenance or surveillance programme[13]. NS-G-2.6 is a basis for the maintenance and surveillance practice at Borssele NPP.

Identification of ageing mechanisms

As a starting point for the AMR for every discipline (mechanical, electrical and civil) catalogues of ageing mechanisms are developed by the OEM (original equipment manufacturer), comprising of all potential applicable ageing mechanisms with detailed information about conditions, location, detection methods etc. The catalogues are made specific for NPP Borssele. Specialists of the plant have reviewed the draft catalogues before the final version was implemented. Every ageing mechanism in the catalogues is supported by several (international) references.

For instance for the ageing mechanism irradiation embrittlement the proceedings of the International Symposiums ASTM STP are used as references. Another example is the use of the proceedings of the two-yearly International Symposium on Environmental Degradation on Materials in Nuclear Power Systems for Environmentally Assisted Cracking (including Stress Corrosion Cracking).

In the AMR a detailed technical evaluation of in-scope SC is performed to demonstrate that the ageing effects will be adequately managed (i.e. the intended function(s) will remain consistent with the NPP licensing basis during Long-Term Operation). The AMR considers the environmental and operating conditions to which each SC is subjected, including system pressure, temperature and water chemistry. These conditions are then evaluated with respect to their effect on applicable ageing mechanisms for each in-scope component.

Once this ageing mechanism evaluation is completed, the necessity for any specific ageing management actions is identified. Effective ageing management may be accomplished by coordinating existing programmes and activities, including maintenance, in-service inspection and surveillance, as well as operations, technical support programmes (including analysis of any ageing

[13] Conceptual document LTO “Bewijsvoering” KCB, NRG-report NRG-22701/10.103460, 9 September 2011

mechanisms) and external programmes, such as research and development. Existing plant programmes and documents are reviewed and evaluated during this step to determine whether existing programmes are adequate without modification, as well as whether existing programmes should be modified. In that case recommendations have been given to implement the modifications.

To give an idea about the information used for the AMR in Figure 2 the contents of a mechanical AMR report are shown.

1 References
2 Abbreviations and Keywords
  2.1 List of Abbreviations
  2.2 List of Symbols and Units
  2.3 List of Relevant Plant ID Codes
  2.4 List of Definitions
3 Introduction
4 AMR Structure and Component Scope
  4.1 AMR Process Scope - Mechanical
  4.2 Ageing Management Review of the SC
5 Current Physical Status of the SC
  5.1 General System Description - Main Coolant System
  5.2 General Description of the SC
    5.2.1 Design Basis
    5.2.2 General Component Description and Function
    5.2.3 System Interfaces
    5.2.4 General Assembly
  5.3 Materials, Design and Specifications
    5.3.1 Pressure Boundary Materials and Design
    5.3.2 Applicable Codes and Requirements
    5.3.3 Safety and Loading Analysis
  5.4 Manufacturing Process
    5.4.1 Semi-Finished Product Documentation and Qualification
    5.4.2 Welding Material Documentation and Qualification
    5.4.3 Welding Processes and Qualification
    5.4.4 Heat Treatment
    5.4.5 Non-Destructive Examination and Testing
    5.4.6 Repair Welds and Non-Conformances
  5.5 Environmental and Operating Conditions
    5.5.1 Design Conditions - SC
    5.5.2 Operating / Service Conditions - SC
  5.6 Plant-Specific Operating & Maintenance History
    5.6.1 Modifications
    5.6.2 Operation and Maintenance History
    5.6.3 Inspection Results
  5.7 Generic Operating Experience
    5.7.1 Generic Nuclear Industry Operating Experience
    5.7.2 Other Relevant Research Results
6 Identification of Relevant Ageing Mechanisms
List corresponding AMs for SC, reference to Catalog of Ageing Mechanisms of Mech. Comp.
  6.1 Thermal Ageing, etc.
7 Identification of Existing Ageing Management Activities
List corresponding AMAs for SC, reference to AMA Notes / LTO Verification of Preconditions (Maintenance, Surveillance, Equipment Qualification, In-Service Inspection & other relevant plant programs ; Identify relevant TLAAs)
  7.1 Monitoring Measures and Programs
    7.1.1 Management of Boric Acid Corrosion
    7.1.2 Water Chemistry Program, etc.
  7.2 Periodic Inspection and Testing Activities
    7.2.1 In-Service Inspection Program (PSI/ISI), etc.
8 Evaluation of Ageing Mitigation Practices during Long-Term Operation
Determine whether existing programs are adequate to manage effects of ageing degradation, also considering relevant operating experience and research results
9 Conclusion

Figure 2 Contents of mechanical AMR report of NPP Borssele

The AMR including the results were reviewed by the Dutch regulator. The AMR was part of the LTO assessment which was used as a basis for a licence change. The licence change was approved[14] in 2013. In the revised licence also requirements are incorporated to enhance the existing ageing management activities based on the results of the AMR and extra recommendations resulting from

[14] Licence: "Wijziging van de Kernenergiewet-Vergunning verleend aan de N.V. Elektriciteits-Produktiemaatschappij Zuid-Nederland (NV EPZ) ten behoeve van verlenging van de ontwerpbedrijfsduur Kerncentrale Borssele (Long Term Operation), DGETM-PDNIV / 13018780, 18 maart 2013"

the regulatory review process. An example of this is a requirement to enhance the existing in-service inspection programme with specified inspections for certain components.

From AMR to a living Ageing Management Process at NPP Borssele

In 2011 the IAEA Safety Guide NS-G-2.12[15] was amended as a guideline within the Dutch Safety Guidelines and became part of the licence of NPP Borssele.

With the results of the LTO assessment (including the AMR) it could be shown that ageing management for all SSC which are important to safety, is in place and adequate. Nevertheless, regarding coordination and traceability further improvement was possible. Therefore it was decided to implement an integrated SSC-oriented ageing management process strongly based on the AMR.

Within the integral management system a specific process was defined, HB-N12-2[16] and PU-N12-50[17] including tasks and responsibilities of the stakeholder within the organization. The scope of the ageing management process is in principle the same scope as the LTO AMR scope. This means that active SSCs are not part of the ageing management process[18]. Ageing of the active SSC is managed by preventive maintenance and or surveillance. This fits to the nature of ageing of passive components for which a coordinated approach is necessary to be able to determine if potential ageing is going on.

In Figure 3 the Ageing Management Process is schematically shown.

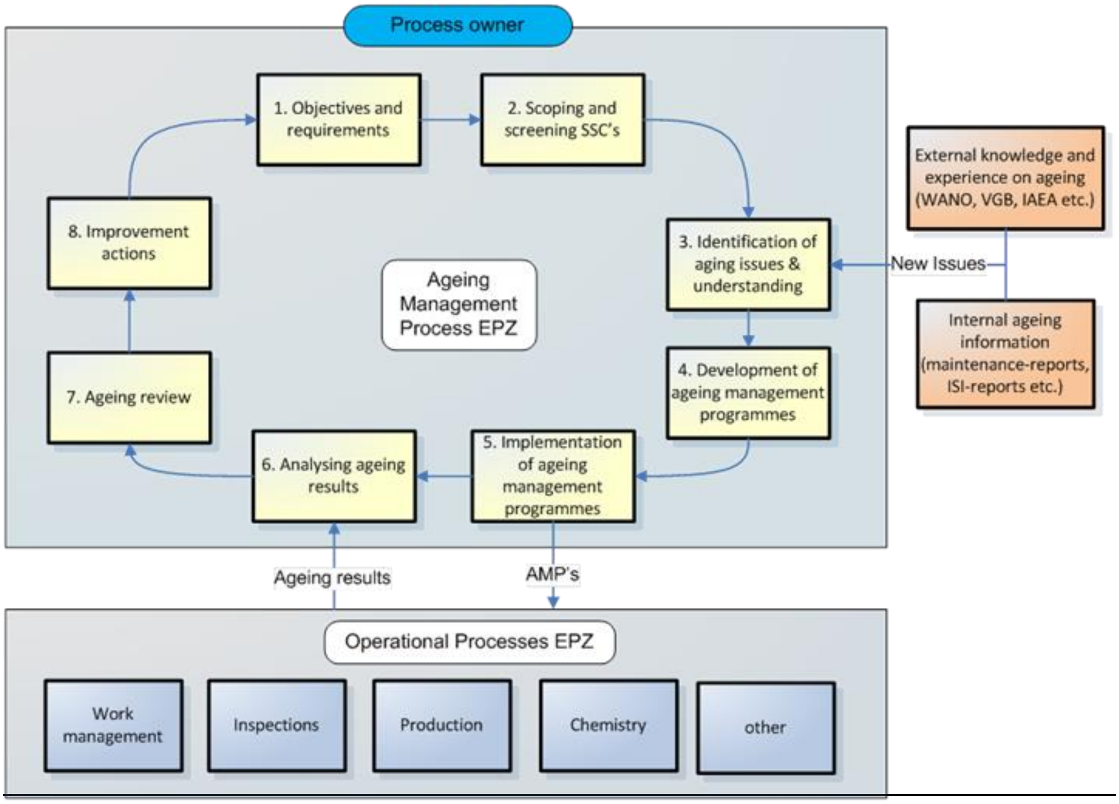


Figure 3 Ageing Management Process at NPP Borssele

[15] Ageing Management for Nuclear Power Plants, IAEA Safety Standards Safety Guide NS-G-2.12, 2009
[16] Handboek Verouderingsbeheersing, EPZ AVS, HB-N12-2, 7-1-2014
[17] Verouderingsbeheer, EPZ Uitvoeringsprocedure PU-N12-50, 5-6-2013
[18] Conceptual document LTO "Bewijsvoering" KCB, NRG-report NRG-22701/10.103460, 9 September 2011

The Ageing Management Team (AMT) is owner of this process and responsible for coordinating the process. The objective is to manage physical ageing in such a way that nuclear safety is maintained throughout the whole lifetime of the plant.

For the in-scope structures and components (no. 2 of Figure 3) all potential ageing mechanisms are identified (no. 3). As mentioned earlier catalogues of ageing mechanisms are used for this including references to international documents including the results of relevant research. Based on that ageing management programmes (AMPs) have been developed and implemented in which relevant activities are identified and integrated to manage these mechanisms (no. 4 and 5). The performance of the underlying activities is the responsibility of the operational departments (schematically shown as 'operational processes in Figure 3). From the ageing management activities results come out and are analysed and reviewed (nos. 6 and 7). This can result in further improvement of the existing ageing management (no. 8). On the right hand side of figure 3 the relation with the existing ageing experience feedback procedure[19] can be seen.

The specific ageing management programmes are developed based on the LTO AMR. In Figure 4 schematically it is shown that both SSC based AMPs and ageing oriented AMPs ('phenomenon based AMPs') are implemented.

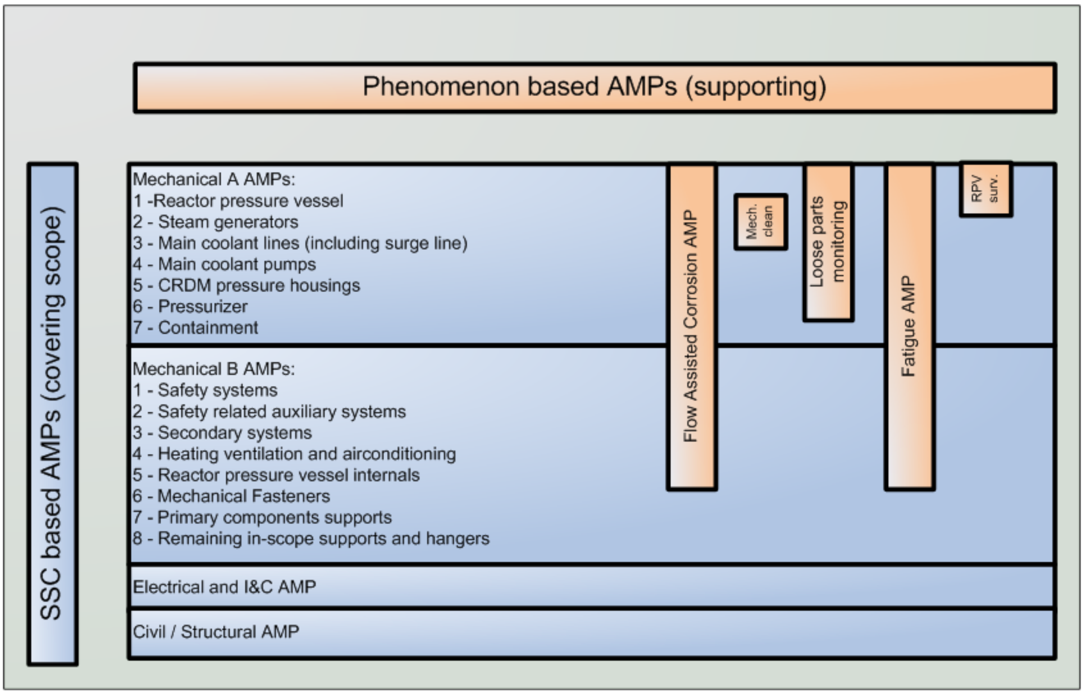


Figure 4 **Matrix structure approach for SSC oriented Ageing Management Programmes at NPP Borssele. Diagram is meant as example; does not give the complete scope of AMPs.**

The format of the AMPs is consistent with the format of IGALL AMPs[20] meaning that in every AMP the nine attributes according to IAEA SRS No.82 are implemented including acceptance criteria and

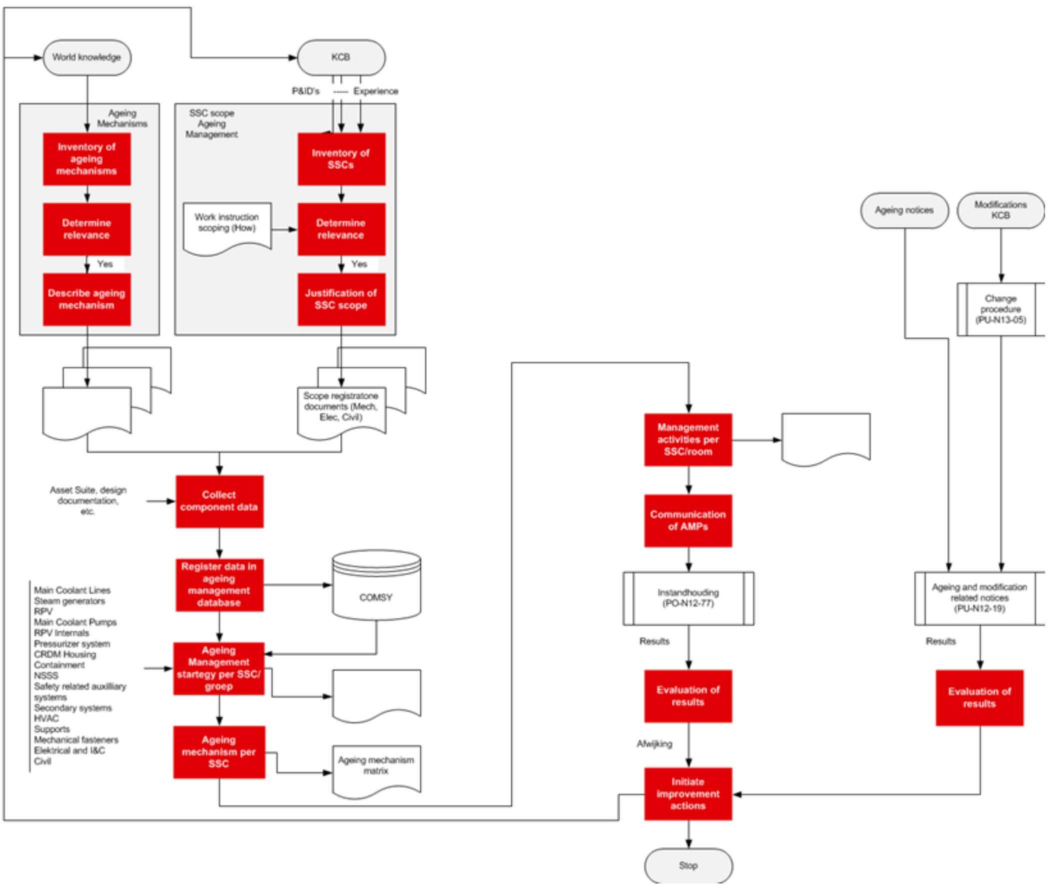
[19] Het analyseren en evalueren van verouderingsmeldingen en van wijzigingsmeldingen, EPZ Uitvoeringsprocedure PU-N12-19, 2-3-2016

[20] Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL), IAEA Safety Reports Series No. 82, 2015

quality assurance. IGALL is also used as a source for enhancing the existing set of AMPs at NPP Borssele. In every AMP reference is made to relevant AMPs of the IGALL database.

In Figure 5 a flowchart is given of the process.

In the upper left part the determination of ageing mechanisms can be seen. The basis is formed by the three catalogues developed for the LTO AMR. These catalogues will be updated in case of new insight comes up for instance as a result of international research or ageing issues at other NPPs. It can also be seen how the scope is determined. Here the scope from the LTO AMR forms the starting point. In case of modifications of the NPP or other reasons the scope can be modified based on a specific instruction. In the left middle of figure 5 it can be seen that data is collected and that all information is available in a specific ageing management database (COMSY, developed by AREVA Germany). Based on the relevant information ageing management strategies are developed. These strategies mostly build upon existing activities in generic programmes like ISI, chemistry etc. The activities themselves are managed and scheduled in the preventive maintenance database. The results from ageing management activities are reviewed and can lead to changes in the AMP. On the right side it can be seen how the existing ageing experience feedback process[21] can lead to changes in the ageing management process. The same is applicable to the modification process which in particular can lead to changes in the scope.



[21] Het analyseren en evalueren van verouderingsmeldingen en van wijzigingsmeldingen, EPZ Uitvoeringsprocedure PU-N12-19, 2-3-2016

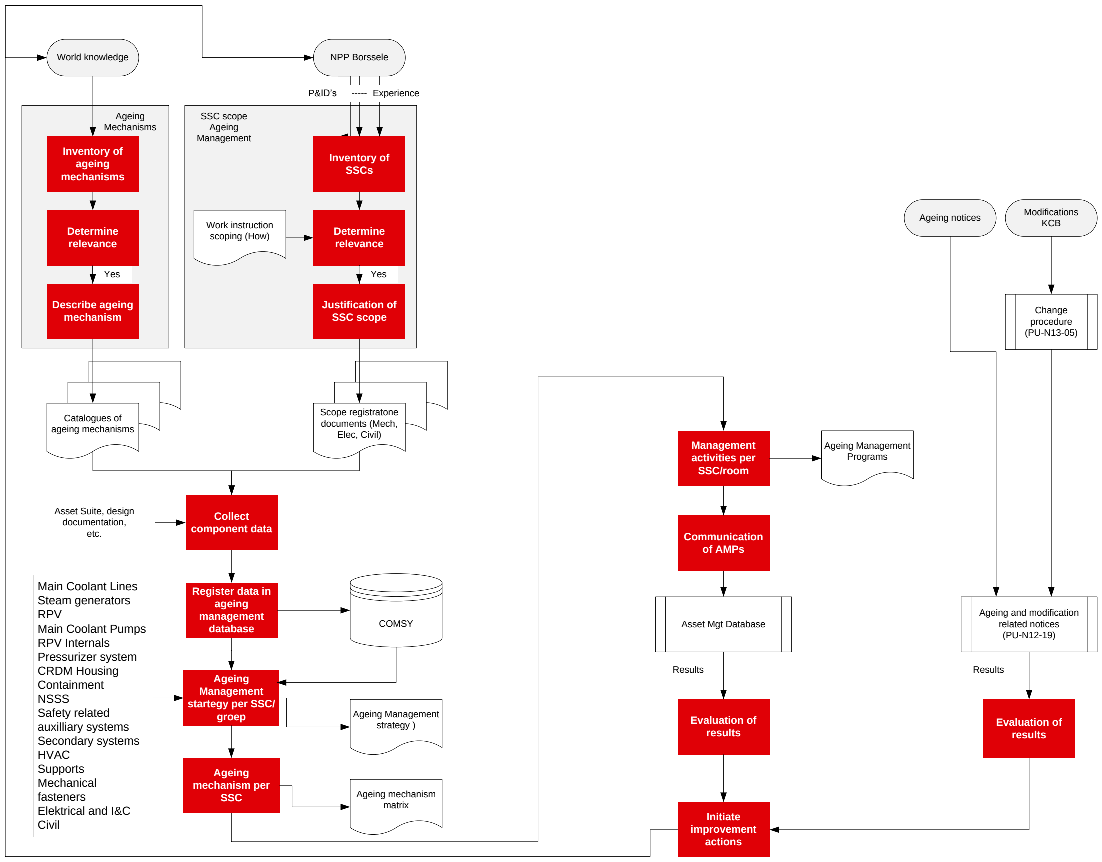


Figure 5 Flowchart of ageing management process PU-N12-50 at NPP Borssele

The AMT is responsible for managing the process but to assure a coordinated approach, working groups are implemented for specific components or commodities consisting of responsible engineers of the involved departments. The working groups are led by members of the AMT. The task of the working groups is to discuss and assess the condition of the specific component or commodities based on the results of the performed ageing management activities. Based on this the working group can propose measures or modify the existing activities. Depending on the impact of the proposal a formal decision process including management approval will be followed to implement the measures or modifications.

2.3.1.2 Scope of the overall AMP - HFR-specific information

Background information on the HFR vessel can be found in Appendix A.

Introduction to overall AMP of the HFR - history

In the Periodic Safety Review (PSR, 10 EVA) performed in 2003-2005, evaluations on ageing were conducted and measures were identified with respect to ageing management and ageing. The evaluation was performed conform IAEA guideline “Implementation And Review Of A Nuclear Power Plant Ageing Management Programme”, Safety Reports no. 15 (1999) and “Ageing Management for Nuclear Power Plants and Research Reactors”, Safety Guide NS-G-382 (Draft, 2005). Only structures, systems and components (SSCs) relevant for safe operation of the HFR were in scope. Through a structured approach, the ageing management activities were mapped and screened for completeness. The measures concluded from the evaluations aimed at a better coordination of the existing activities (organization). Among the measures were for instance an NDT inspection on possible pitting of the steel containment dome (no degradations were found) and to implement a monitoring programme on the alignment of buildings.

At the HFR, upon request of the Dutch regulatory body (as required by the HFR-licence), three Integrated Nuclear Safety Assessments for Research Reactors (INSARR) were conducted by the IAEA (in 2005, 2011 and 2016).
In the 2005 mission, upon request of NRG and JRC (Joint Research Centre, the owner of the HFR), special attention was paid to the subject ageing. The INSARR team had two recommendations on the maintenance procedures for electrical equipment. The recommendations were the addition of a regular schedule (expiration dates) on the preventive and periodical maintenance procedures (PPOs) and to file information of electrical equipment per piece of equipment (analogous to the PPOs and files on mechanical SSCs). The INSARR team suggested to establish an integrated maintenance management system, with a standard and regular collection of equipment data (one system instead of two). In the 2011 mission again special attention was paid to ageing as a part of maintenance management. There were two recommendations with respect to ageing, one on limiting the leak of the pool and one on a feasibility study on creating a watertight room below the reactor tank (sub pile room). In the 2016 mission on the subject ageing, again a recommendation was issued related to the leakage of pool water. The IAEA recommended to investigate leak paths again and implement corrective actions in order to limit the leak rate. Refer to section 2.5.2 for NRG's follow-up actions. Some other recommendations are indirectly related to AM. One of them being the recommendation to recruit a Maintenance Manager, a position which was not really filled for about three years. This has been fulfilled beginning of 2017. The routine at the HFR of conducting every 10 years a Periodic Safety Review (10 EVA) including ageing management with respect to safety in the scope was considered by the IAEA as a good practice. In the future the INSARR missions will be followed up after two (this time in 2018) instead of 5 years.

In 2019/2020 an IAEA safety review mission to the HFR focused on ageing management will be organized. The IAEA has developed a specific methodology for safety review on ageing in research reactors called CSO (Continued Safe Operation). The methodology is comparable to the methodology for power reactors (SALTO) but tailored to the characteristics of a research reactor.

In recent history at the HFR three mayor events occurred which have a relation with ageing management e.g. the BPL (bottom plug liner), the BPS (bottom plug seal) and the corrosion of a buried pool water transport line between buildings.

No. 1, Bottom Plug Liner, BPL

During a visual inspection deformations were observed in a small area in the primary outlet lines embedded in the 2m thick concrete floor of the reactor pool of the HFR. The deformations were observed during a specific non-routine visual inspection and acknowledged as a relevant deterioration in 2005. Studies and observation during repair revealed that deformations were caused by galvanic corrosion driven by reinforcement steel unintentionally in contact with the aluminium primary outlet lines. Thus, the deterioration was caused by a construction error which took more than 40 years to reveal itself. With inspections at locations in the installation where no deviations are expected it is possible to detect ageing problems as a result of construction errors. In 2010 the lines were repaired in a complex repair procedure reconstructing the affected areas of the primary outlet lines embedded in the 2 m thick concrete floor of the reactor pool.

No.2, Bottom Plug Seal, BPS

During a maintenance stop it was observed that there was a very small leak path between the primary system and a part of the pool water system. Investigations revealed a very small gap in a large seal (ring shaped seal of the order of 1 m diameter) between the liner at the bottom of the pool and the bottom plug which guides the control rods through the pool floor into the reactor tank. The cause of the small gap in the seal was never established with certainty. There are no signs of deterioration of the seal or the seal surfaces. It is not excluded that the seal was not perfectly leak tight since the construction of the reactor. A small design modification was carried out to solve the problem.

No.3, Tritium

In September 2012, the groundwater monitoring system around the HFR revealed unexpected high levels of Tritium in groundwater near the HFR. Investigation revealed a small leak in a buried pipe used during maintenance stops for transport of pool water to storage tanks set in an adjacent building. The leak was caused by galvanic corrosion between a steel strap around the aluminium pipe wrapped in bitumen. Deterioration of the bitumen layer enabled the contact between the strap and the pipe. The pipe was replaced by a above ground pipe surrounded by another pipe with leak detection. It was then not covered in the HFR maintenance programme, but today all relevant buried pipes are covered in a monitoring programme. The event has been published to the parliament, presented to the IRSRR meeting and listed in the national report of the Convention on Nuclear Safety, where the Netherlands includes the HFR information on a voluntary basis.

Current ageing management at HFR covers a strategy including preventive and corrective measures taken in order to detect degradation of systems structures and components (SSC's) and ensure they remain within acceptable limits. An effective ageing management programme will contribute to availability and reliability of safety critical SSC's. Understanding of the installations relevant degradation mechanisms is key to development of an effective ageing management programme.

NRG develops its ageing management programme further based on two condition assessment programmes executed between 2013 and 2014. The first consisted of a installation integrity scan of the HFR. Hence this programme was called Asset Integrity Programme (AIP). The AIP resulted in a list of installation modifications which are executed in the years since 2014. In addition the AIP stated the need for an additional condition assessment related to ageing. As a result an ageing management review (AMR) was executed in 2014.

The purpose of the AMR was to determine whether the actual ageing management measures granted sufficient ageing control of safety critical SSC's. The review consisted of a breakdown of necessary measures compared to measures used within the HFR maintenance and operation plus identification of probable gaps. Identification of the ageing control measures is part of ageing management improvements. The IAEA recommended approach was used as guidance for development and implementation.

The AMR was a scan of the HFR ageing conditions at a fixed time. Recommendations have led to both installation improvement and development of new continuous improving ageing programme. This programme consists of extension of the existing preventive and corrective maintenance concept and a maintenance workflow system suitable to maintain and improve the developed concept. The programme concept will be registered in the NRG management system. The improvement of this maintenance programme has started in 2016 and will be challenged by the 2019/2020 SALTO-mission. The recommendations of the SALTO assessment will be implemented accordingly.

Actual scope of the AMP at the HFR

The AMR and recommendations were delivered by a temporary project team of specialists. The recommendations have been handed over to the HFR engineering and maintenance department. High priority recommendations have already been executed. Further development of the AM programme is assigned to the manager maintenance.

The HFR safety relevant systems and structures are listed below. Furthermore, auxiliary systems are identified. Based on the listed systems, the related safety critical systems combined with safety critical components subject to the AMR are defined. The safety critical SSC's are SSC's of which failure may result in a loss of safety function of a system. Such failure may either occur in the system the component belongs to or another system. In both cases the component is safety critical. In the first review of the AMR all components were considered to be safety critical. A second stage review has excluded all components that are not safety critical.

List of systems:

1.) Shutdown system
    a. Regulating rod control
    b. Lower and upper regulating rod guidance and guidance fixation to vessel
    c. Regulating rod guidance and transmission on bottom plug
    d. Regulating rod drive system below bottom plug
2.) Primary system with reactor core vessel
3.) Pool cooling system
4.) Off-gas system
5.) Ventilation system
6.) Instruments
7.) Buildings
    a. Reactor containment building
    b. Reactor detached building/ office
    c. Primary pump building

d. Ventilation building (incl. ventilation shaft)

e. Emergency power supply building

8.) Emergency power supply facilities

9.) Special services

a. Jacket pipe drying system

b. Jacket pipe leak detection

c. Hot Cell

10.) Instrument air system

11.) Compressed air system

12.) Pool experiment cooling water system

13.) Drain system

a. Warm drain

b. Hot drain

c. Ion drain

d. Contaminated drain to waste pit

e. Pool liner drain

f. Waste water line to DWT

g. Water tanks near decontamination

14.) Cranes and containers

The SSCs have been divided into several groups. SSCs of a particular group are those components manufactured from similar materials or having similar working principles within one system like cooling water pumps and valves.

In addition a list of components based on the IAEA SSG-10 expected to be relevant to ageing has been created. The components identified during the AIP are specified on this list too.

Continuous monitoring of the condition of the SSCs will be managed by the extended preventive maintenance programme. Development and implementation of this programme started in 2016.

Methodology of AMR for safety critical systems

This part describes the methodology used for selection of the AMR of safety critical systems.

The ageing management programme consists of:

- Determine applicable AMR SSC’s scope;
- Identify and understand relevant ageing mechanisms;
- Prevent or minimize ageing;
- Detect, monitor and execute trend analysis of ageing mechanisms;

- Mitigate ageing mechanisms;
- Establish continuous improvement of the ageing management programme;
- Record results.

AMR data is collected by execution of the first four steps mentioned above. Based on the collected data the current condition is compared to the required condition. The methodology used is described in this section.

Safety critical functions and components

Only safety critical SSCs are subject to the AMR which complies with the IAEA SSG-10 guideline. All safety critical SSCs of the listed systems are identified. This process is illustrated by Figure 6.

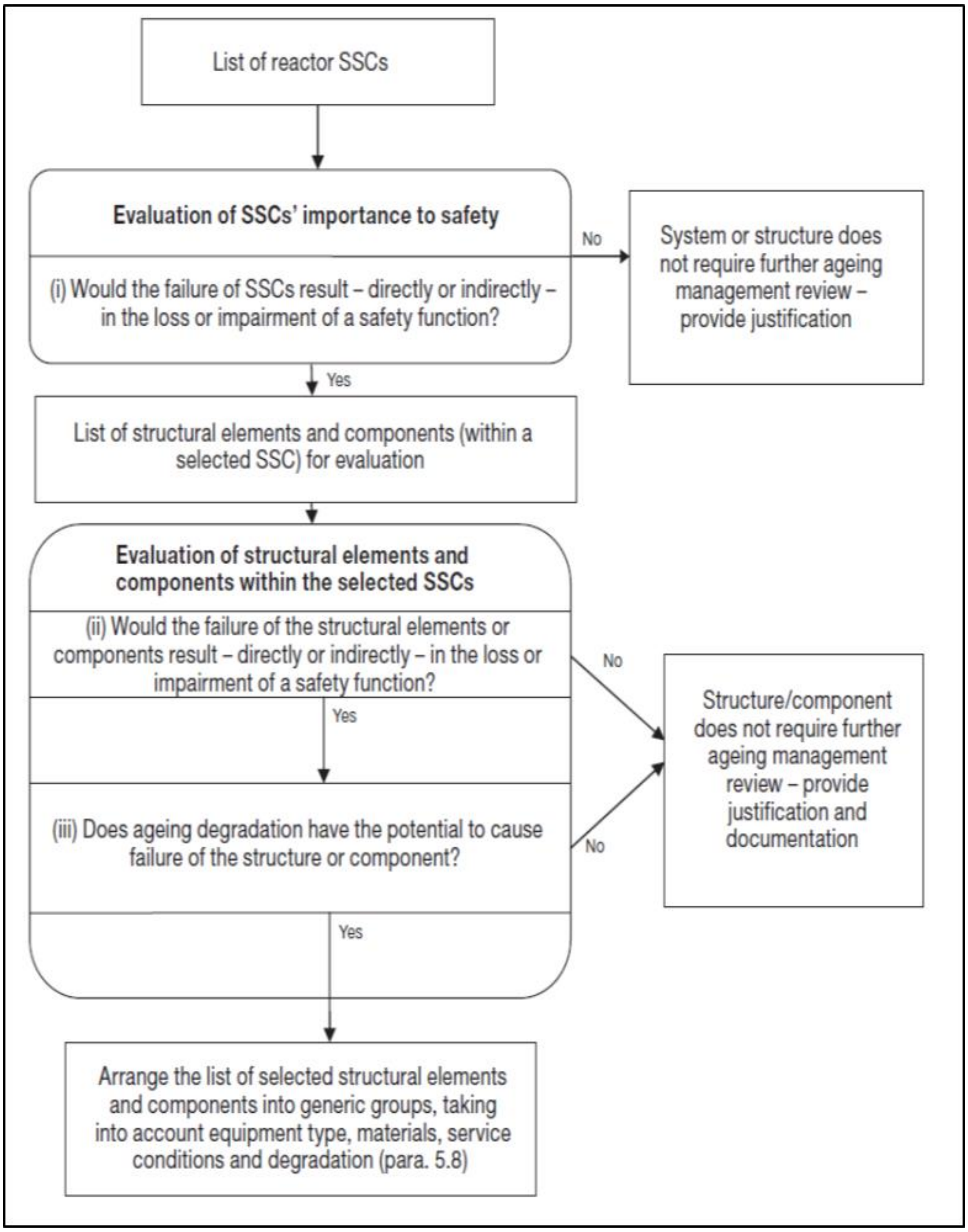


Figure 6 SCC identification at the HFR

Common cause failure of ageing effects are taken into account when selecting relevant systems and components. The common cause failure may reduce the effectivity of redundancy or diversity. Hence redundancy or diversity is not considered to result always in a required safety level. In practice this may lead to a redundant system to be part of the AMR as a common cause failure may lead to a failure of both systems at one time.

Identification of safety critical relevant SSCs

A safety critical SSC is defined as an SSC which is needed to support one of the three basic safety functions. The HFR safety functions are defined as:

- Safe reactor shut down during all operation conditions and after all possible emergency conditions;
- Cooling of fuel elements during all operation conditions and after all possible emergency conditions;
- Containment of radioactive substances by remaining safety barrier integrity.

The safety functions may be detailed for each specific safety critical system. The safety critical supporting auxiliary systems need to be taken into account too.

Quality assurance and AMP future development

The AMR is a screening on a fixed moment. Therefore also a programme to ensure continuous AMP improvement had to be established for the HFR. This programme was initiated in 2016 by defining maintenance procedures. As from the second half of 2017 the development of the HFR maintenance concept and service providing has been started. New procedures on service providing will be managed by the existing NRG management system.

The basic outline of the programme will be:

- Development of a reliability centred maintenance (RCM) programme (maintenance concept)
- Additional long-term operation (LTO) review
- Service providing programme consisting of:
    - Workflow process for corrective maintenance;
    - Failure report and corrective action system (FRACAS) for continuous maintenance concept improvement;
    - Computerized maintenance management system (CMMS) for maintenance programme and action registration.

2.3.1.3 Scope of the overall AMP - HOR-specific information

Responsibilities at HOR

The AMP is embedded in the operations department of the HOR reactor. The structure and the responsibilities of this department are described in the document with the title 'Beschrijving KEW vergunningsgebonden organisatie'. In this document, it is secured that the director of the Reactor Institute Delft has the mandated responsibility for the operation, maintenance and safety of the reactor on the short- and long-term conforming to legal obligations. The head of the operations department is in charge of execution of the daily activities related to this responsibility.

Despite the fact that some reactor related SSCs are embedded in the 'Facility Management & Vastgoed' (FMVG) organisation, the above stated document secures that the head of the operations department is responsible for the reactor-related activities of the people working within this part of the FMVG organisation.

Within the operations department the function of ageing management coordinator is performed by the staff member of the department. The central role of the ageing management coordinator can be seen from the figure below, which depicts the roles and responsibilities of the various parties involved in the ageing management programme.

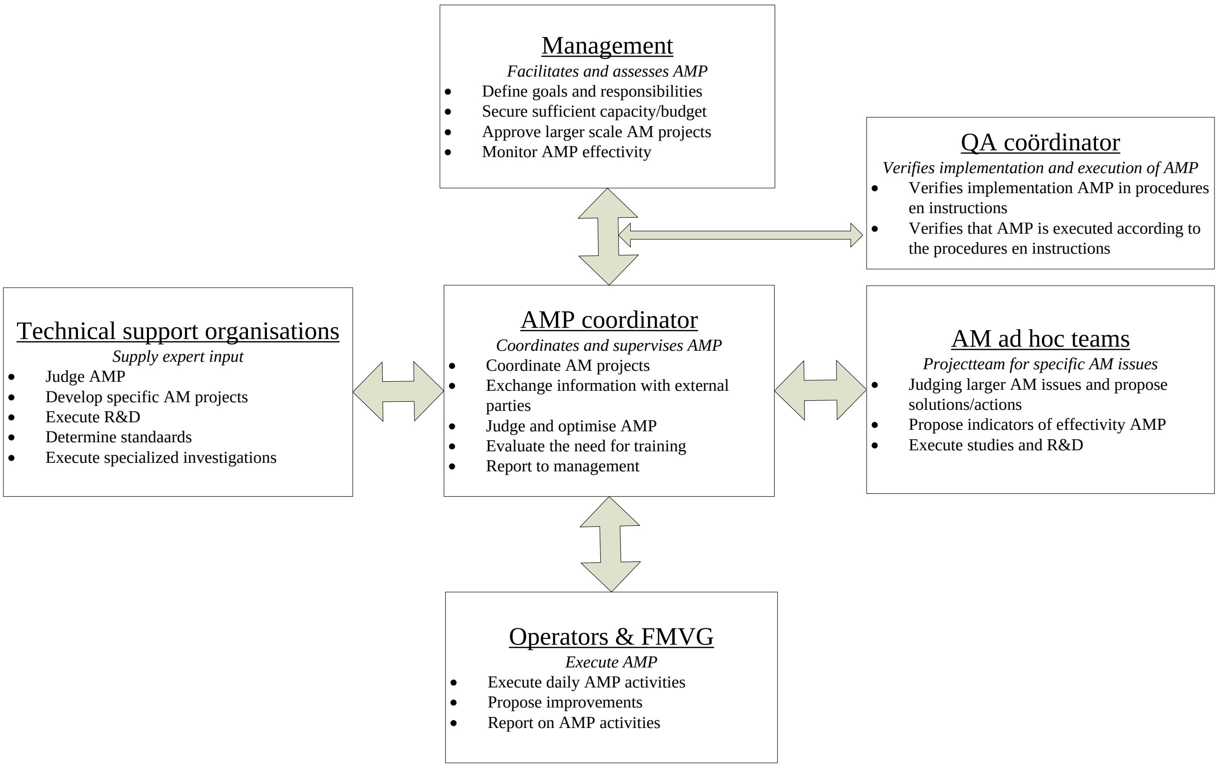


Figure 7 Identification and grouping of SSC's into functional units within the scope of the overall AMP at RR HOR

Following the guideline as provided by the annex II of the IAEA SSG-10 as a starting point and using multi discipline input from all involved parties in facilitating the operation, maintenance and inspection of the HOR reactor, all SSCs as present were systematically identified and grouped into functional units. Special focus was given to the SSCs in the highest safety class by extending the ageing analysis down to the component level for those SSCs. For SSCs in a lower safety class the ageing analysis was performed on the functional level, finishing the analysis on the system or structure level depending on what is the functional unit.

The three main safety functions of any reactor are:

1. Safe shutdown

2. Sufficient cooling
3. Isolation of the reactor environment from the ambient environment in case of an accident.

Safety classification of SSCs is chosen accordingly. From a safety perspective, inspection and maintenance activities should focus on maintaining these functionalities. The importance of the SSCs for safety defines the safety and determines the focus in a safety orientated maintenance strategy.

Developing appropriate ageing management programme projects at RR HOR

Based on the identified possible ageing degradation mechanisms (see also section 2.3.2.3 'Ageing assessment at the HOR' for more details on this process) per functional unit (SSC), it was evaluated if existing maintenance and inspection activities and related instructions can and will identify the ageing mechanisms under consideration. It was evaluated if the instruction defines the acceptance criteria well enough and if the record keeping of the instruction is clear enough to assess ageing degradation over time. The related HOR instructions, check sheets and fill out forms have been evaluated based on these criteria to perform the initial gap analysis. To fill in the gaps in the existing maintenance and inspection activities, an ageing management project list has been composed. Projects on this list have been prioritised based on a score for the various effects that can occur, when the SSC fails due to the ageing mechanism under consideration. The list of effects not only includes the safety class affected of the SSC, but also effects like non-availability, environmental effect, non-compliance to licence conditions, etc.

Now that a project list has been defined based on the gap analysis and has been prioritised based on the effect list, the approach during the execution phase of the AM projects is to use the Reliability Centred Maintenance (RCM) methodology. RCM is characterised by the following:

- RCM is driven by safety. Economic considerations are only of secondary importance. Safety has to be guaranteed. If safety is not compromised any appropriate way of maintenance can be chosen based on economic grounds.
- RCM focuses on maintaining the functionality of the functional unit. Individual components might fail as long as the functionality of the unit is maintained. It should be realised that maintenance cannot improve the safety of a functional unit as compared to the design. Maintenance can only maintain the same safety level as incorporated during design.
- RCM is a continuous process. Differences in expected (compared to actual) life times and ageing mechanisms need to be identified and investigated.

The RCM method at RR HOR consists of the following steps:

1. Collect relevant information and (operational) data on the SSC under consideration;
2. Perform RCM analysis;
   a. Follow the below RCM logic tree;
   b. Determine the various options for maintenance activities and intervals;
   c. Estimate the effectiveness and costs;
3. Document the outcomes of the RCM analysis and chose the maintenance tasks;
4. Evaluate the maintenance by analysing the operational data and external information.

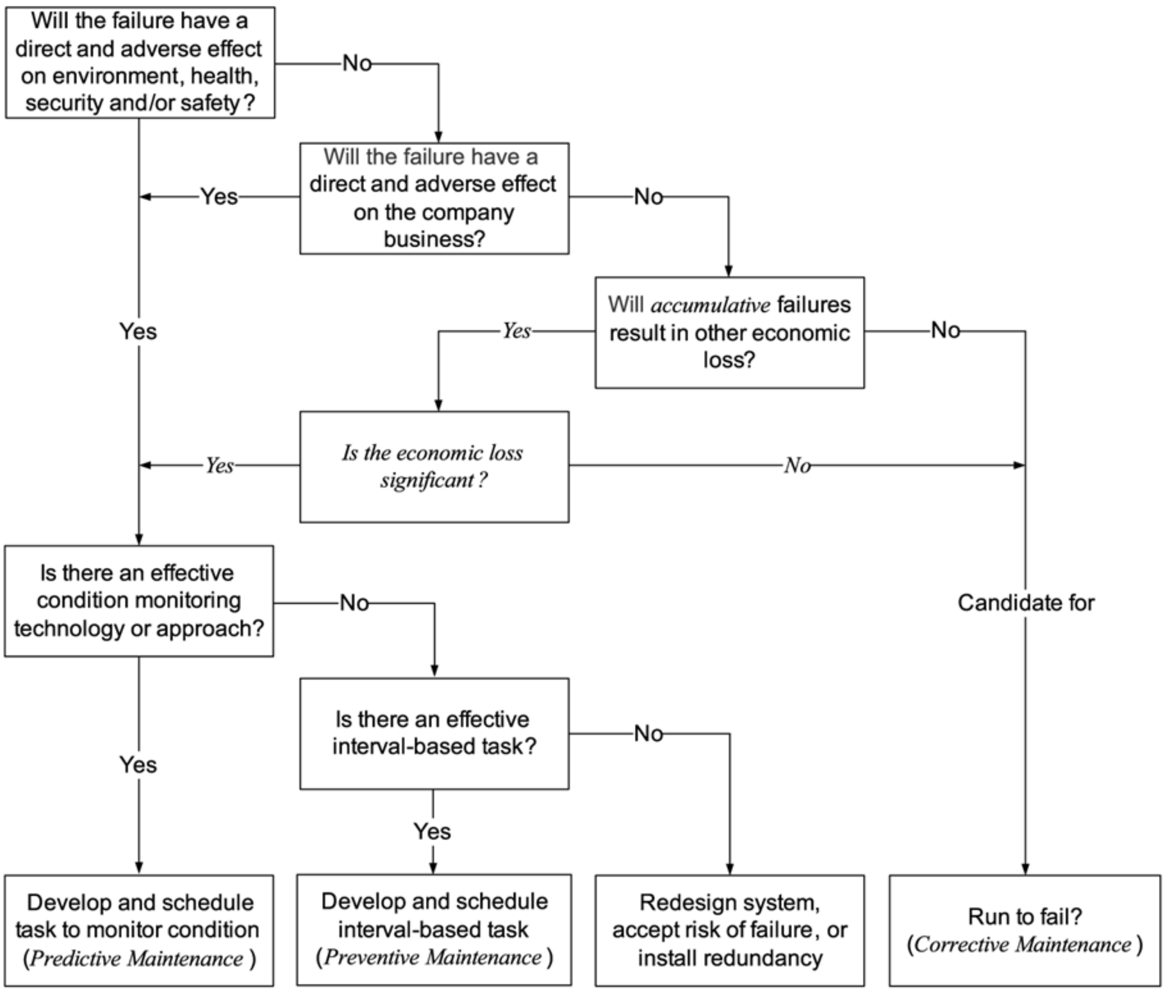


Figure 8 Reliability Centred Maintenance tree of RR HOR

To guarantee that also in the future the maintenance and inspection activities on the various SSCs will be identifying the relevant ageing mechanisms, the procedure as described in the previous paragraphs is formalised for newly identified ageing mechanisms and/or SSCs in the procedure HOR2016-014P “Ageing management” and related instructions. Such newly identified ageing mechanisms (and resulting AMP activities) can be initiated at any moment, when new insights and/or information (internal or external) becomes available.

The overall ageing management programme itself is evaluated every 10 years in the frame of the 10-yearly Periodic Safety Review (10EVA). The requirement for a 10-yearly evaluation is formalised in the European Union’s Nuclear Safety Directive 2014/87/Euratom, but has already been a licence requirement for the HOR since 1996.

With the introduction of the RCM methodology the effectiveness of the maintenance activities (as they have evolved over the operational years of the reactor) should be analysed. To be able to do so operational data shall not only be evaluated on the short term (hour-to hour, week-to-week, etc.), but also over the long term. To facilitate this process one of the major focus points in setting up the ageing management programme is to bring together the required long-term operational data in one digital database. Long-term trend analysis on the selected data has now become possible and limits for these trends have been determined. Focus on adapting the maintenance activities shall be on those SSCs that show rapidly changing trends.

2.3.2 Ageing assessment

2.3.2.1 Ageing assessment - NPP Borssele

Ageing assessment has been addressed in section 2.3.1.1 'Scope of the overall AMP & Ageing management- NPP Borssele'.

2.3.2.2 Ageing assessment - HFR-specific information

Effective ageing management requires a systematic approach. The systematic approach as proposed by the IAEA is given in Figure 9. The process loop was originally designed for power reactors, however is applicable to research reactors as well. The loop represents a Deming-cycle. The approach consists of 5 elements: 1) "understanding", 2) "plan", 3) "do", 4) "check" and 5) "act". The loop is not sequence dependent. The separate elements however may influence each other. Tests for instance may show an ageing profile more progressive than expected which may lead to modified preventive maintenance and inspections of the ageing management programme ("plan"). A change of understanding as well as results out of steps 3 and 4 may lead to changes in all elements (understanding, plan, do, check and act). For the AMR information out of all steps is needed to assemble a complete overview of ageing management measures and gap review of existing measures.

During the AMR the process loop was used for the ageing condition assessment and determination of actions to be taken. The AMR process was a first step executed only once. The ageing management programme to be used today, which is being further developed, is described in sections 2.3.3.2 'Monitoring, testing, sampling and inspection activities - HFR-specific information' and 2.3.4.2 'Preventive and remedial actions - HFR-specific information'.

<u>Element 1 - Understanding</u>: "understanding ageing of a SSC"

The first step is to understand the mechanism by asking questions like:

- What is ageing;
- Which ageing mechanisms may occur, which of these mechanisms are relevant and what effects are caused by these mechanisms. In other words understanding of the SSC ageing process (5.9 of IAEA SSG-10);
- How could the ageing be prevented or limited (5.12 of IAEA SSG-10) and if necessary be mitigated (5.23 of IAEA SSG-10);
- Where can ageing occur;
- How can ageing be determined (5.14 of IAEA SSG-10) and if necessary be mitigated (5.23).

In addition two types of time related effects need to be taken into account: physical ageing for example by corrosion and non-physical ageing for example by out-of-date (obsolescent) components for which no spare parts are available anymore (3.1 of SSG-10). For the ageing management improvement the IAEA approach covers safety critical SSCs only. The approach for screening of SSCs is shown in Figure 9 (taken from SSG-10).

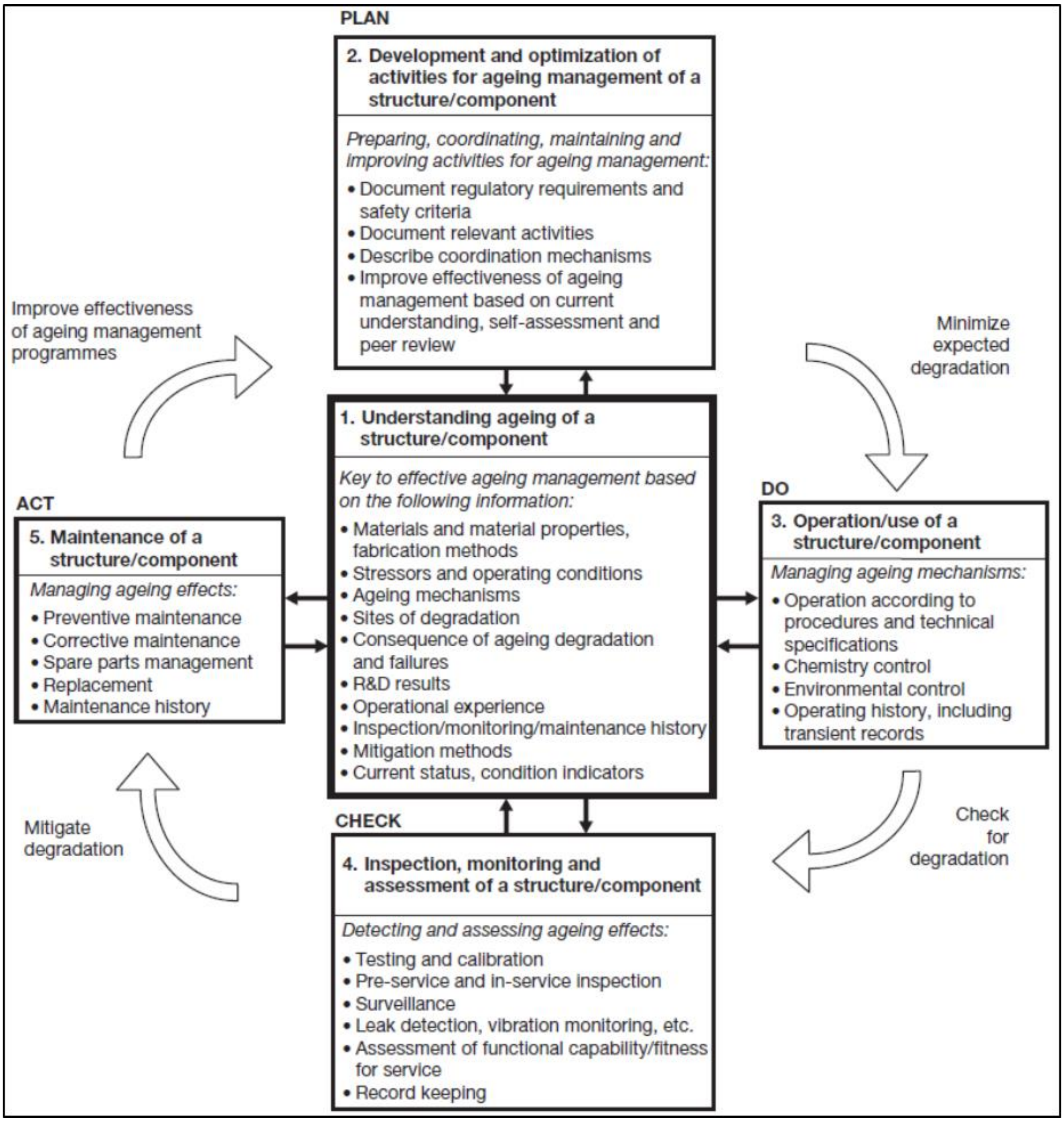


Figure 9 **Process loop of ageing management control at the HFR**

Element 2 – Plan: "Development and optimization of activities for ageing management of a structure/ component"

Once understanding the installation's specific ageing mechanisms, a plan for managing these ageing mechanisms has to be established (for both physical and non-physical effects). This includes an evaluation of the existing methodology and practice used within the HFR organization to prevent and mitigate ageing of components in which operational experience and (internal and external) research results are taken into account. The following methodologies and practices are being evaluated at the HFR (5.23 of SSG-10):

- Maintenance strategy;
- Overhaul of SSCs;

- Modification of SSCs;
- Adjustments of operation conditions;
- Other practices which may influence ageing.

Element 3 – Do: "Operation/use of a structure/component"

The conditions under which an SSC is operated, influence the ageing of its components. Relevant conditions are (3.14 of IAEA SSG-10):

- Material stress;
- Temperature;
- Pressure;
- Chemical condition;
- Operation conditions like, radiation, humidity, occurrence of active chemical liquids and gases;
- Wear as a result of usage, which includes modification of dimensions and relative positions of parts and assemblies.

Element 4 – Check: "Inspection, monitoring and assessment of a SSC" (refer also to 2.3.3.2)

Components which are subject to ageing mechanisms must be monitored. The following research methodologies may be used to identify such components:

- Performance test: functionality of a SSC;
- Periodical functional test: ability of a SSC to perform its function;
- Inspections: Periodical condition of the SSC is being inspected and recorded;
- Monitoring: Continuous SSC monitoring by means of direct or indirect process condition parameter monitoring which gives information about the SSC;
- Non- destructive tests like visual, surface, volumetric and ultrasonic inspections;
- Leak detection and vibration measurement, visual inspection and in service inspections.

Inspection activities which should be executed at the HFR are (5.18 of IAEA SSG-10):

- Condition observation of SSC's (e.g. leakages, sounds or vibrations); usually managed during periodical reactor walk down;
- Sampling of water coolant for chemical and/or radiochemical analysis.

Ageing effects may be detected by a change in measurable operating parameters (e.g. control rod drop time, water chemistry parameters, temperature, flow rate and pressure). Parameters that can be predictive of ageing degradation should be routinely monitored (either on-line or periodically). Readings should be assessed and trends determined, in order to predict the onset of ageing degradation in a timely manner.

Element 5 – Act: Maintenance of a SSC (also refer to section 2.3.4.2, 'Preventive and remedial actions - HFR-specific information').

Assessment execution

The ageing assessment is executed according the process loop described above. In the assessment the following topics have been considered:

- Materials;
- Key ageing mechanisms;
- Conditions;
- External sources;
- Existing R&D programmes.

Materials

Within the HFR several materials are being used. To determine the SSC's materials several sources have been used:

- Valve list: a list of all mechanical components which specifies used materials;
- DOKPAK (in depth technical description of the installation);
- Knowledge of technical experts, asset register in SAP and P&IDs.

The data of above mentioned sources is generated from manufacturer and designer data.

All materials are indexed in different groups. A conservative approach is used for ageing mechanism determination. If needed a more detailed approach is used for specific components. This approach may be continued for future investigation and execution of ageing management.

The conservative approach is based on the worst case scenario of conditions. Even if it is extremely unlikely a certain ageing condition will be met this conditions is considered to occur.

Key ageing mechanisms

The ageing mechanisms are defined according the IAEA SSG-10, annex 1:

1. Change of material properties due to neutron radiation;
2. Change of material properties due to operation temperature;
3. Stress/ stress relaxation or creep (as a result of mechanical/ thermal stress and operation temperature;
4. Motion, fatigue, or wear (as a result of altering temperature, flow, load and vibration);
5. Corrosion and oxidation processes;
6. Chemical processes other than oxidation or corrosion;
7. Erosion;
8. Changing technology;
9. Changing regulation;
10. Out dated documentation.

Conditions

Conditions are both operation conditions caused by processed matter and environmental conditions the components face.

Operation and environmental conditions are (physical parameters):

1. Temperature of processed matter;
2. Pressure of processed matter;
3. Flow of processed matter;
4. Chemical conditions of processed matter;
5. Radiation;
6. Environmental temperature;
7. Environmental pressure;
8. Environmental composition (e.g. air or concrete);
9. Cyclic changes of operation parameters.

Comparable conditions are grouped in order to establish a consistency in ageing mechanisms determination of combined materials.

The AMR covers a technical analysis of the installation. The process is focused on physical ageing mechanisms (1 to 7). As long as the existing technology remains sufficient to manage ageing processes this technology will be qualified as sufficient. This condition however will only be applicable as long as spare parts of the existing technology remain available. To ensure the ageing management process remains compliant an evaluation of regulation compliancy and technology is made each 10 years (10-yearly periodical safety review).

External sources

Besides the use of internal data external sources have been used for improvement of the HFR AMP.

NRG contributes to the IAEA Incident Reporting System for Research Reactors (IRSRR) and relevant reports from other reactors are monitored. At the HFR every 5 years an Integrated Safety Assessment of Research Reactors (INSARR) is conducted. Ageing management is part of the scope of the assessment.

NRG has set up a cooperation agreement meeting with the Opal reactor in Australia and the Safari reactor in South Africa.

Existing R&D programmes

Apart from the AMR and further programme development the following ageing monitoring programmes are already being executed:

- Surveillance Programme (SURP); Core vessel material assessment against design specifications (running since 1984 when the vessel had been replaced) supervised by Lloyd's Register. In the SURP programme the change of the vessel material properties (fracture toughness) due to irradiation with neutron are determined. In this way it is assured that the material properties are present as assumed in the design of the vessel and necessary to withstand fracture growth.
- In Service Inspection programme (ISI): Component condition assessment with respect to the welds and wall thickness of the core vessel and primary cooling water lines according to the ASME XI standards. In the in-service-inspection of the welds in the vessel and the primary piping,

the size of defects are reported against criteria on the critical fracture length adopted. NRG develops its own dedicated inspection equipment.

2.3.2.3 Ageing assessment - HOR-specific information

The ageing mechanisms taken into consideration for RR HOR are taken from the annex II of the IAEA SSG-10 and are the following:

1. Change of material properties due to radiation;
2. Change of material properties due to temperature;
3. Creep of stress due to pressure;
4. Movement, fatigue or wear due to temperature of flow variations;
5. Corrosion;
6. Change of material properties due to chemical processes;
7. Erosion;
8. Change in technology;
9. Change in regulation;
10. Incomplete documentation.

As can be noted from the last three bullets above, obsolescence is thus included in the IAEA guidelines, while explicitly excluded from the Technical Specification for the National Assessment Reports.

For the identified SSCs the relevant ageing mechanisms were determined using historical data (including operational data as well as manufacturing data), experience (as formalised in the maintenance instructions), manufacturers information (manuals), engineering judgement and international databases for ageing. The IAEA Research Reactor Ageing Database (RRADB) and the IAEA Incident Reporting System for Research Reactors (IRSRR) were already systematically reviewed for applicable information during the 10-yearly safety reassessment. The ageing management analysis of an international peer was also used as input to determine the relevant ageing mechanisms for the respective SSCs.

The HOR does not participate in external ageing research projects (these are mainly focused on NPPs), but receives information on the outcomes of such programmes through the following research reactor networks, which meet on a yearly schedule:

- Research Reactor Operators Group (RROG) and
- ‘Arbeitsgemeinschaft für Betriebs- und Sicherheitsfragen an Forschungsreaktoren‘ (AFR).

In both network groups ageing management is a continuously reoccurring item as most research reactors in Europe have been in operation for several decades.

Based on the experiences and input from the above sources, the existing instructions for inspecting and maintaining the various SSCs at the HOR were evaluated with a focus on acceptance criteria. The majority of the projects executed during the initiating phase of the ageing management programme (2012-2016) had as a goal to determine the acceptance criteria for the SSC where quantitative acceptance criteria were not yet determined. The outcomes of these projects were used to update the relevant instructions with appropriate acceptance criteria.

2.3.3 Monitoring, testing, sampling and inspection activities

2.3.3.1 Monitoring, testing, sampling and inspection activities - NPP Borssele

The ageing management activities at NPP Borssele are part of several programmes which come together in a generic asset database[22] which forms the input for work preparation, planning, executing, recording and trending of the activities. In Figure 10, a schematic overview is given of the different generic programmes, their basis (regulation) and the connections. This structure includes all activities necessary to bring and keep the plant in the technical condition required to fulfil its functions.

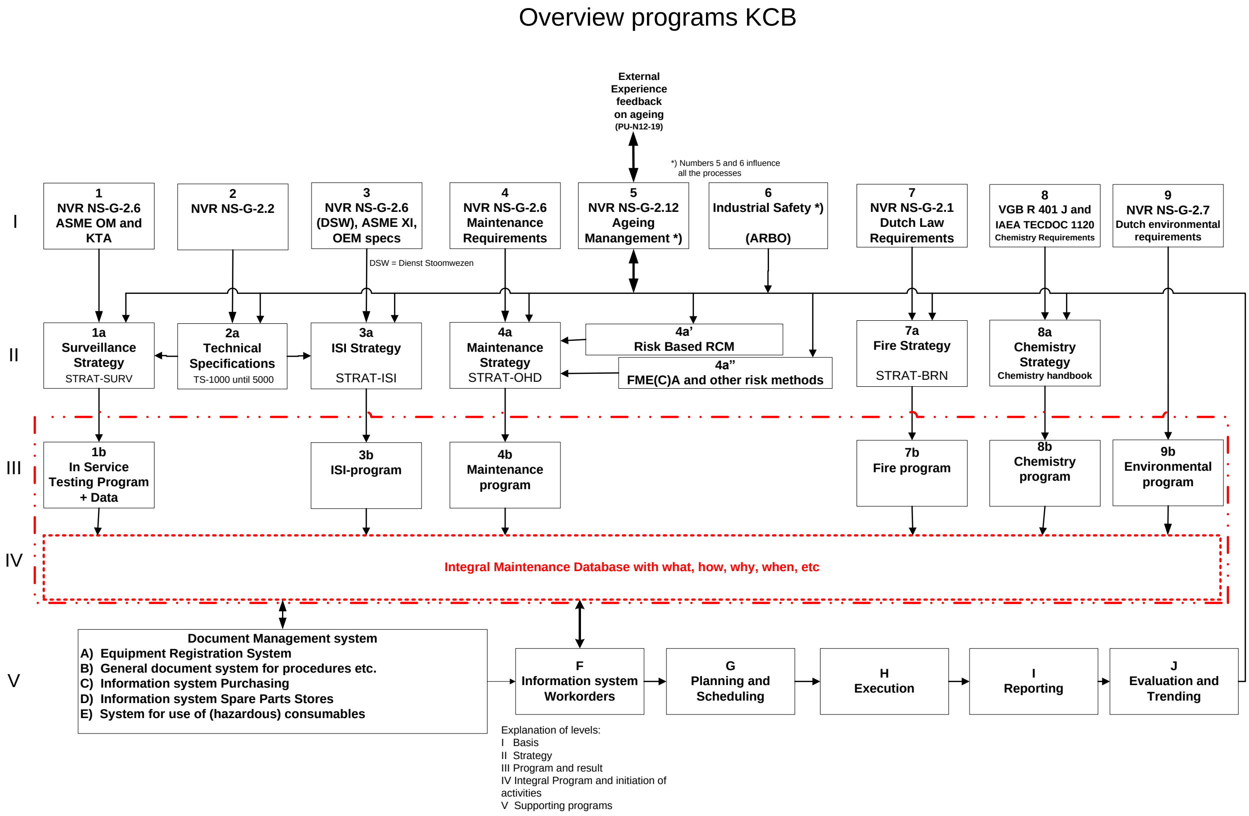


Figure 10 Schematic overview programmes in place at NPP Borssele

For some ageing mechanisms specific monitoring programmes are in place.

An example is the Fatigue Loads Monitoring programme. In this programme fatigue loads are monitored and reported every year to check if the occurred loads are still in line with the assumptions of the fatigue analyses. The assumed loads are written in a load catalogue. For specific locations thermal transients can give substantial thermal fatigue loading. For these locations FAMOS (AREVA product) thermocouples are in place to be able to measure detailed temperature fields including stratification effects. These data are monitored continuously and yearly assessed and if needed used to change the operation procedures to reduce fatigue loads.

[22] Dutch: 'Instandhoudingsdatabase'

Another example is the Surveillance Programme for Neutron Induced Embrittlement. This programme comprises of specimens of the beltline materials which are irradiated in the RPV during operation. The original programme with three sets of specimens was enhanced with three extra sets of specimens because of the extension of the operation lifetime from 40 to 60 years of operation. The last specimen set will be taken out in 2018.

Leakage monitoring is very important to have early awareness of active corrosion processes. On the one hand leakage detection is in place to identify critical leakage in a timely manner to ensure that adequate corrective measures can be taken. Leak detection in larger areas is generally aggregate (i.e., accumulation from a selected group of rooms or areas) and not restricted to specific rooms or systems. On the other hand walk downs are performed to visually identify leakages.

The In-Service Inspection programme is very important to assure the integrity of structures and components of the NPP. By means of qualified volumetric, visual and/or surface examinations SCs are tested.

The ISI-programme at NPP Borssele is based on the ASME XI guidelines and requirements of the Dutch Law for pressure equipment enhanced with inspections based on operating experiences and regulatory requirements. Based on the Ageing Management Review for Long Term Operation also extra inspections were introduced partly one-time and partly added to the programme.

Nowadays NPP Borssele follows a ten-yearly inspection interval meaning the timeframe in which all required inspections have to be performed. The current interval ranges from 2010 until 2019 and is based on ASME XI edition 2007.

All the inspections are performed by certified inspection firms. Based on the Dutch law the nuclear regulator (ANVS) has delegated the surveillance (including qualification) of the approved ISI programme to an independent certified third party. Lloyd's Register is acting as authorized inspection agency for nuclear pressure equipment in The Netherlands.

In case of findings not fulfilling the acceptance criteria, at least a condition assessment has to be performed. Further operation with or without repair has to be justified in a written document and approved by the regulator.

The Water Chemistry Programme at KCB is an existing programme that monitors and adjusts the chemistry properties in the Primary Systems, Secondary Systems, Nuclear Safety Systems, Safety-Related Auxiliary Systems, and in the Heating, Ventilation and Air-Conditioning Systems. The programme relies on monitoring and control of main coolant water chemistry based primarily on the VGB guidelines for primary and secondary water chemistry, as well as external experiences (e.g., from EPRI, EDF, and WANO).

2.3.3.2 Monitoring, testing, sampling and inspection activities - HFR-specific information

In order to determine the condition of safety critical SSCs a number of the methodologies are being used.

- Visual inspections which are part of the inspection programmes and planned preventive maintenance orders (PPO);
- Pre Operational Checkout procedures.

SURP

The reactor vessel material properties are monitored by means of the surveillance Programme (SURP). The programme describes periodical mechanical testing actions of irradiated vessel material samples and is supervised by Lloyd's Register, the notified body for nuclear pressure equipment in The Netherlands.

ISI

The reactor vessel with its primary cooling water pipelines, connected pool penetrations and systems directly related to the three main safety functions are inspected according the in service inspection programme. The inspections include surface and volumetric inspections (for instance ultrasonic, eddy current and visual). The scope of the inspection programme is based on analyses of remaining safety function in case of leak. This includes the vessel and all primary cooling waterlines from the reactor vessel to the primary heat exchangers. Also this programme is supervised by Lloyd's Register.

The inspection results will be followed up by corrective maintenance actions if needed.

PPO

Results of inspection related planned preventive maintenance orders (PPOs) are used for corrective and improvement actions. The current PPOs have a weekly, monthly, yearly, 3 yearly or 6 yearly frequency.

This total maintenance programme will be reviewed and extended based on a reliability centred maintenance approach in 2017 and 2018. The new programme will include more preventive activities used to monitor the ageing processes of the HFR and the application of a more structured approach with acceptance criteria, indicators for trending and effectiveness assessment.

Pre operational checkout

After a reactor shutdown before start up pre-operational checkout procedures are executed. The pre-operational inspection consist of about 40 procedures as listed in the pre-operational inspection index. The procedures contain instructions and check lists used for both operational activities and installation condition inspections. An inspection can be a visual and functional of the related system.

2.3.3.3 Monitoring, testing, sampling and inspection activities - HOR-specific information

In this section, a general overview is provided on the monitoring, testing, sampling and inspection activities as performed at the HOR to identify unexpected behaviour or degradation during service. As nearly all operational activities and measurements are performed with the above goal it is not feasible to describe all these activities in detail here. The below description therefore focuses on the activities related to maintaining the three main safety functions; safe shutdown, sufficient cooling capacity and isolating the reactor environment from the ambient environment in case of an accident.

Testing activities at RR HOR

The HOR reactor instrumentation has to be functionally tested each and every time before operating the reactor, as stated in the operational licence. This functional testing of the instrumentation and other essential SSCs is captured in the so-called checkout instruction. Since the HOR runs a weekly cycle (continuous operation from Monday till Friday, except for scheduled maintenance periods during the summer and winter holidays), this results in a high frequency of testing the correct functioning of the reactor instrumentation and other essential SSCs.

Upon completion of the reactor operational period on Friday evening the manual reactor shutdown buttons are used to stop the reactor and the time it takes for the control rods to drop is automatically determined. Hereby functionally testing the safe shutdown functionality at least every week.

More rigorous instructions for testing and calibrating each and every measuring channel of the HOR are available and performed with a periodicity as specified by the instruction. Both of the above testing activities can only be performed when the reactor is not in operation. Thus, leaving Monday mornings and the maintenance stops as times to perform these tests.

Monitoring activities at RR HOR

Both monitoring on component level and on functional level will be discussed in this section. Monitoring activities on components relevant to safety will be discussed first. For the description of the monitoring activities on the functional level, the continuous monitoring systems on safety functions will get focus.

<u>Monitoring degradation of components</u>

Surveillance activities during operation of the reactor

During operation of the reactor the operator group performs three hourly plant walk down rounds based on a dedicated instruction. These three hourly plant walk downs serve multiple purposes (nuclear and conventional safety, health physics, operational and logistical items, etc.). Amongst those purposes is ageing management.

The results of these surveillance activities are documented in a standardised worksheet. This worksheet has been adapted several times based on the findings of the Ageing Management Programme.

Inspection and sampling activities when the reactor is not in operation

During the times when the reactor is not operated more elaborate inspections are executed based on a periodic test list. Examples are measuring the thickness of the control rods and the time it takes for the control rods to drop from their highest position to guarantee safe shutdown. Also the locations which are not accessible during operation of the reactor due to increased radiation levels (like the piping corridor) are inspected during these times. The results of these inspections are captured on standardised worksheets. During the initiating phase of the ageing management programme relevant information from these inspection records was selected for trending and a dedicated digital database was made to facilitate this trending.

Sampling taking of water for offline sampling is normally also performed when the reactor is not in operation. Water from the beam tubes is checked offline on conductivity and basin water is checked offline for nuclide content.

To fill in the gaps as identified during the initial gap analysis when developing the HOR ageing management programme (see section 02.3.1) quite a few items were added to the periodic test list.

As these inspections all have their own intervals the periodic test list has evolved from a paper check list to a digital database, which gives notifications when an periodic inspection is upcoming.

Third party inspections on the reactor containment

Third party inspections on the containment have been executed every two years since the completion of the reactor containment building in 1963. Any changes to this construction have to be notified in advance and inspected upon completion by the third party inspection company. In the past third party inspections were performed by the governmental organisation for supervision on pressure and boiler vessels (the so called ‘Stoomwezen’). Due to changing responsibilities for supervision of pressure retaining vessels by the introduction of the European Pressure Equipment Directive (PED in short) and the explicit exclusion of nuclear pressure equipment in this directive, a separate nuclear pressure directive (the so called regeling nucleaire drukapparatuur) is implemented in the Netherlands. Supervision of nuclear pressure vessels is now performed by authorized inspection agencies. The ANVS authorizes and supervises these inspection agencies.

Due to this shift in responsibilities the inspection programme for the containment is currently being updated with more quantitative acceptance criteria by the HOR.

The updated programme will be reviewed by the third party inspection company and will be offered to the ANVS for approval. Besides the two yearly inspection of the reactor containment to identify degradation of the containment, a functional leak test is performed every year for the containment. This leak rate test is witnessed by the third party inspector once every four years. Lloyd’s Register is acting as authorized inspection agency for nuclear pressure equipment in The Netherlands.

In accordance with the Dutch nuclear pressure directive the containment is the only pressure equipment inspected by the authorized inspection agency. With regard to ageing degradation there are no requirements for further third party inspections at the HOR.

Monitoring safety functions

Continuous condition monitoring on safety functions is performed by the reactor instrumentation. Process conditions and safety conditions are monitored by separate systems, as dictated by the defence in depth concept. For both systems alarm levels are set. Some selected abnormal process conditions can result in an automatic run down of the control rods, but most process channels only generate an alarm when a threshold level is reached. It is up to the operator to respond to such alarms. Reaching an alarm level on a safety channel results in automated corrective action. This can be a shutdown of the reactor, isolation of the reactor containment and or isolation of the pool, depending on which safety channel (or combination of these safety channels) has reached the alarm level.

A safety function is created by SSCs making up the safety function. Thus by monitoring the safety function it can be concluded that the SSCs, making up the safety function, are providing the function, as they should. Therefore a selection of the monitoring actions by the reactor instrumentation can be related to ageing degradation. Examples are described below.

Example: Water level monitoring

The water level in the reactor basin is monitored continuously by redundant and diverse measurements to ensure sufficient cooling capacity is available for the reactor core. For the HOR no active cooling is needed after shutting down the reactor. Thus, by ensuring sufficient water is present in the basin it is guaranteed that sufficient cooling capacity is available. The temperature of the basin water is maintained below certain values during operation. Next to monitoring the water level in the basin, also any leakages through the basin lining are collected in two small volume collection tanks. At the top of these collection tanks a high level detection is available. By the above measurements, the functional integrity of the pool is monitored.

Example: Radiation monitoring

Radiation monitoring aims at detecting unacceptable ageing degradation of the fuel cladding and shielding.

An extensive monitoring programme is available for detecting fission products both in the water and the air just above the basin. Monitoring for fission products in the water is done both online and offline (sampling) and by a diverse set of detectors and installations. Any rise in the signals as detected by these installations can be an indication for a degradation of the fuel cladding.

Area radiation monitoring is also available within the reactor containment building to detect any increased radiation within the reactor building. Area radiation monitoring is performed independently by the instrumentation of the HOR operations group and the instrumentation of the radiation protection department of the Reactor Institute Delft.

2.3.4 Preventive and remedial actions

2.3.4.1 Preventive and remedial actions - NPP Borssele

During the operating life of NPP Borssele both preventive and corrective measures haven been taken because of ageing issues. In such cases specific procedures are followed to assure that the measures are performed properly and do not jeopardize the structural integrity of the NPP SSC's.

Some examples of preventive and remedial actions are mentioned below.

Steam Generator Tubes

In the early years of operation NPP Borssele followed a phosphate secondary water chemistry regime. One of the results was uniform corrosion (wastage) of some of the Incoloy 800 tubes of the two steam generators. This was managed by plugging the involved tubes. In the eighties the copper condenser tubes were replaced by titanium tubes making it possible to change to a high AVT (All Volatile Treatment) water chemistry and a PH > 9.7. Because of this the wastage of the tubes was stopped. In the early nineties again some tubes were found (eddy-current testing) with decreased wall thickness because of fretting damage. After investigation it was found that this was a generic design issue also applicable for some other KWU NPPs. The tube supports were modified to incorporate special comb-shaped tube supports in the U-bend region of the bundle supports. Fretting damage was mitigated by this modification. In recent years some tubes in the hot legs were found with single axial indications just above the tube sheet pointing to a stress corrosion cracking mechanism. After a comprehensive assessment by the NPP, the OEM and peer reviews by independent specialists it was concluded that the existing sludge on the tube sheet is responsible for a local corrosive environment. In 2017 both mechanical and chemical cleaning was performed to reduce the corrosive environment. In addition Film Forming Amines have been injected. By both preventive and remedial actions the NPP is managing the integrity of the steam generator tubes for 60 years of operation.

Qualification of Design Base Accident Resistant Electrical Equipment

Electrical equipment is identified which has to perform its safety functions during and/or after design base accidents. For this equipment a programme is in place (explained in chapter 3) in which the qualified residual life is determined every year based on monitoring of the ageing stressors temperature and irradiation. In case of a qualified residual life less than 5 years the relevant components have to be replaced or a (specific) qualification programme has to performed to prove that the qualified residual life is sufficiently long. Particularly in the period 2013 – 2016 several replacement projects have been performed based on this programme and some qualification programmes. EPZ works together on this with German and Swiss NPPs within the VGB.

2.3.4.2 Preventive and remedial actions - HFR-specific information

During the life time of the HFR the reactor vessel has been exchanged. In order to monitor the condition of the replaced vessel in time, SURP has been established. In this programme samples of material equal to the core vessel construction material are placed at various positions in the reactor. The samples are inspected and analysed during the running time of SURP. Results will be used to determine remedial actions on the core vessel if needed.

During reactor operation, when the irradiation rig on the central PSF (pool site facility) position is moved outwards from the irradiation position close to the vessel core box, the rig does no longer shield the thermal neutrons scattered back by the pool water towards the vessel core box. This results in an increase of the fluency on the outside of the vessel core box material. For this reason, during reactor operation, when a rig is moved outwards, a special shielding plate is positioned against the core box outer wall to avoid unnecessary irradiation of the vessel core box by thermal neutron returning from the pool towards the vessel. This prevents unnecessary ageing of the vessel core box material.

The pool leakage rate is monitored to enable the reduction of risks from presence of pool water at unanticipated locations in the installation.

In the secondary cooling water system after erosion was detected after bends and valves due to cavitation, erosion monitor plates are mounted in the cooling water line.

Contact pressure indicators in the system showed deterioration due to vibrations. These switches were replaced with glycerine filled ones. The pressure connection tubing has been replaced by coiled tubes.

2.3.4.3 Preventive and remedial actions - HOR-specific information

Preventing favourable conditions for ageing

Besides the testing, monitoring, inspection and off-line sampling activities to detect degradation (as generally described in the previous section on the HOR, 2.3.3.3), degradation is minimized by applying high quality grade materials when designing. During operation, favourable conditions for ageing are avoided by controlling the operational environment within stringent margins. Especially for an open pool type reactor with a relative low thermal power the operational environment can be considered as mild with regard to operational radiation levels, operational pressures and temperatures.

As an important example of actively avoiding favourable conditions for ageing of the components within the basin, the water treatment installations can be considered. Water quality is continuously controlled and monitored by the water treatment installations. By controlling the water quality (pH and conductivity) ageing of the components in contact with water is minimized. Not only the water quality in the primary circuit is being controlled and monitored, but also the water in the secondary circuit is controlled and monitored, but within less stringent margins. Other examples of avoiding favourable conditions for ageing are the shielding of radiation sensitive SSCs and avoiding rapid temperature changes in both the containment and the basin.

Preventive maintenance

If a SSC is selected for preventive maintenance based on the criteria as shown in Figure 8 an effective time interval shall be determined for the SSC. At the HOR most of the effective time intervals for preventive maintenance are historically determined, based on over 50 years of operating the reactor. For new components selected for preventive maintenance the advice of the manufacturer is requested on an effective time interval. If little or no information is available and the application is critical, endurance tests can be performed or a very conservative time interval can be selected. Replaced items are always inspected for signs of ageing degradation and the effective time intervals for preventive maintenance can be adapted based on to the outcome of the inspections of the used items/parts.

The effective time intervals for the various SSCs are captured in a digital database, which gives a timely notification to prepare the preventive maintenance action.

Remedial actions

If a SSC is selected for preventive maintenance or if despite the above activities, a SSC fails or if a SSC is selected for corrective maintenance and fails, a remedial action is needed to repair the functionality of the failing SSC. Such a remedial action normally requires a new/spare component and an instruction for the replacement and verification of the correct functioning of the SSC.

During the initiating phase of the ageing management programme a new structure was set up for spare part management. This includes a new database structure for keeping track of the availability of spare parts, but also set up the programme for actively checking the correct functioning of the safety essential spare parts. For the verification of the correct functioning of the SSC after the repair the normal instruction for testing and calibrating the SSC can normally be used. If no such instruction is available a dedicated inspection and test plan shall be written in advance and be verified by the head of operations.

2.4 Review and update of the overall AMP

2.4.1 Review and update of the overall AMP - NPP Borssele

The ageing management process PU-N12-50 is part of the Integrated Management System (IMS[23]) which is under control by the Quality Assurance department. This department performs audits. In case of minor or major conformities the process owner is responsible for implementing improvements.

In- and external ageing events are assessed by the AMT according to a specific procedure and can lead to changes in ageing management activities on a technical level. In case of plant modifications the scope (SSCs) of the overall ageing management programme will be changed.

As can be seen in Figure 3 and has been described in 2.3.1.1 and 2.3.2.1, a closed loop is in place in which the effectiveness of ageing management is measured and assessed.

As a part of the LTO assessment (described in NRG-22701/10.103460[24]) Time Limited Ageing Analyses (TLAA) have been revalidated. These TLAAs comprise of four groups of analyses:

- Reactor Pressure Vessel embrittlement (fracture mechanics and fluency calculations)
- Fatigue (cumulative usage factor including environmental fatigue)
- Leak Before Break (fracture mechanical calculations)
- Environmental Qualification of accident resistant electrical equipment (residual lifetime calculations based on temperatures and radiological parameters)

If applicable current “state-of-the-art” including R&D results are taken into account in ageing management. Periodically this will be done by the 10-yearly Periodic Safety Review (PSR) but also in between sometimes “state-of-the-art” is taken into account. A strong mechanism for this is the international network in which staff of NPP Borssele is in. One of the networks is within the German VGB in which working groups are in place on themes related to ageing. Besides NPP Borssele and all German utilities also Swiss NPPs are involved and in some groups also EDF is participating. Also important is a strong relation with the OEM of the NPP: AREVA NP Germany. In case of specific issues it can be decided to engage for further R&D on specific topics. In several cases working together with foreign NPPs or other organizations is done.

In this respect it is important to mention that the German phase-out of Nuclear Energy will mean than the network of the German NPPs (via VGB or directly) will vanish in the near future. Until 2023 the German utilities will still be involved in the network although the financing of joint research projects has decreased significantly. Because of the commonalities in design, working together with German NPPs is very fruitful for NPP Borssele. One measure for Borssele is to look for more strong cooperation with still operating KWU NPPs outside Germany. Also in working together on relevant projects or issues the focus will be more on the rest of the nuclear world. An example is the RPV embrittlement research project CAMERA. This project is a continuation of the projects CARISMA and

[23] Handboek Integraal Management Systeem, EPZ AVS, HB-A-00, most recent update of 25-09-2017
[24] Conceptual document LTO “Bewijsvoering” KCB, NRG-report NRG-22701/10.103460, 9 September 2011

CARINA in which NPP Borssele and all the German NPPs participated via the VGB. In CAMERA NPP Borssele is involved together with a Swedish and a non-KWU Swiss NPP.

Engineers of NPP Borssele are involved in several international groups and conferences and have a lot of contacts with engineers at other NPPs.

Every 10 years a comprehensive PSR is performed according to the recent IAEA standard on Periodic Safety Review, which is currently the IAEA Safety Standard Series No. SSG-25[25]. As can be seen in the safety factors also ageing and related topics are part of the scope of the PSR. Based on the outcome, after every PSR a follow-up plan is drafted with proposals for improvements. After approval of the plan by the regulator, EPZ is committed to implement the improvements. In the last PSR at NPP Borssele (2010-2013) credit was taken on the results of the comprehensive LTO assessment[26]. The assessment and results were reviewed by the Dutch regulator. The AMR was part of the LTO assessment which was used as a basis for a licence change. The licence change was approved in 2013[27]. In the revised licence also requirements are incorporated to enhance the existing ageing management activities based on the results of the AMR and extra recommendations resulting from the regulatory review process.

In the upcoming PSR ageing management will be reviewed again.

Unexpected or new issues are incorporated in the AMP by running the process as described in Figure 3 and Figure 5 including the possible input from external events according to the ageing experience feedback procedure[28].

2.4.2 Review and update of the overall AMP - HFR-specific information

The gaps found during the AMR are improved by means of a deterministic approach on the SSCs for which gaps are found. Control steps are defined to improve the ageing management control programme. Need, extend, frequency and method depend on the consequence of failure.

The result of this approach will lead to an identified inspection programme. This programme is combined with a reliability centred maintenance (RCM) programme. The improved ageing management programme is implemented in the maintenance programme. Based on a failure mode effect and criticality analyse (FMECA) method the RCM maintenance programme is being maintained.

This maintenance programme is managed in a computerized maintenance management programme (CMMS) for which a SAP software module is used. The CMMS contains the HFR asset register. All ageing management related preventive maintenance and inspection programmes are managed by means of asset register allocated programmes. The result of inspection and possible corrective actions is allocated to the related asset register location as well. All the asset register locations holding safety critical SSCs will be identified by a safety critical SAP code.

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[25] Periodic Safety Review for Nuclear Power Plants, IAEA Safety Standard Series No. SSG-25, 2013

[26] Conceptual document LTO “Bewijsvoering” KCB, NRG-report NRG-22701/10.103460, 9 September 2011

[27] Revised licence: ‘Wijziging van de Kernenergiewet-Vergunning verleend aan de N.V. Elektriciteits-Produktiemaatschappij Zuid-Nederland (NV EPZ) ten behoeve van verlenging van de ontwerpbedrijfsduur Kerncentrale Borssele (Long Term Operation), DGETM-PDNIV / 13018780, 18 maart 2013’

[28] Het analyseren en evalueren van verouderingsmeldingen en van wijzigingsmeldingen, EPZ Uitvoeringsprocedure PU-N12-19, 2-3-2016

The results of inspection programmes and corrective actions on SSCs marked 'safety critical' will create an understanding of the programme efficiency. If the programme as developed and improved is found to be insufficient, a programme modification will be made. This is the HFR maintenance workflow.

The maintenance workflow loop will be further developed at the HFR. The basic loop is shown in Figure 11.

In the second half of 2017 NRG started with the development steps for this maintenance workflow loop. This development consists of the following:

- Complete review and extension of the existing maintenance concept and maintenance plan;
- Implementation of improved maintenance workflow which includes a workflow for corrective and preventive maintenance and a failure report analysis and corrective action system (FRACAS);
- The maintenance concept will be reviewed by means of a maintenance failure mode and criticality analysis (FMECA) methodology. The most critical safety and reliability critical installation parts will be reviewed with priority, hence the first part of the concept will be available for use by the end of 2017;
- The Safety Assessment on Long Term Operations (SALTO) of 2019/2020 and the follow-up in 2022 will be used to assess the recommendations delivered by the established programme.

At the same time the first steps are being taken for implementation of an improved workflow system which will eventually generate the conditions for a FRACAS managed maintenance concept.

The complete maintenance workflow as registered in the CMMS will generate an overview of the HFR ageing condition. The data collected to generate this overview consists of trend analysis and PPO inspection results.

PSR

In addition to the ageing management programme improvement, a IAEA SSG-25 based periodic safety review (PSR) is executed by NRG inspection services. This review is executed every 10 years (see introduction in section 2.3.1.2).

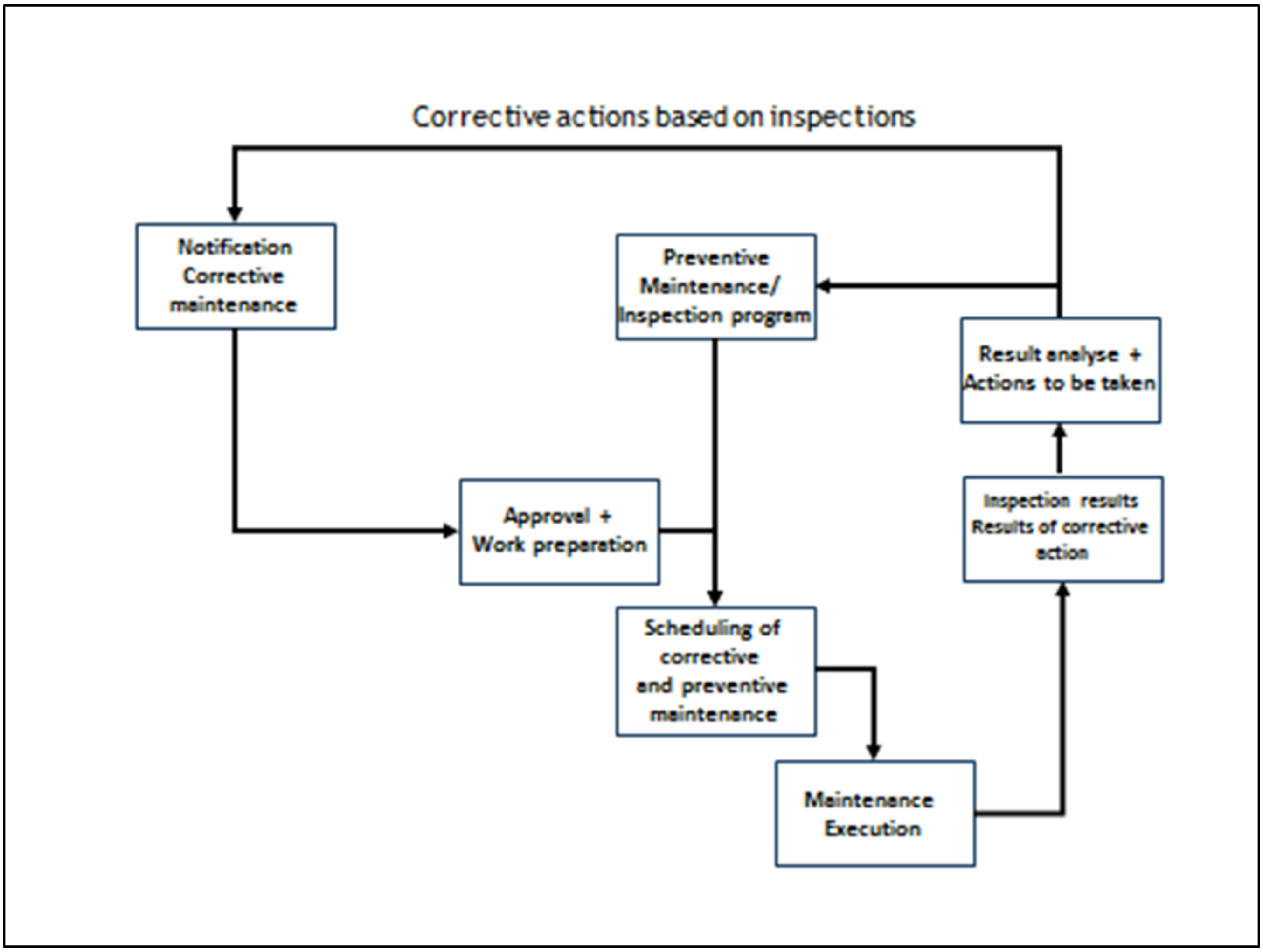

Figure 11 Maintenance workflow loop at the HFR

2.4.3 Review and update of the overall AMP - HOR-specific information

Continuous improvement

New or unexpected ageing issues are continuously incorporated into the AMP by the ageing management procedure (HOR2016-014P) that was developed, based on the experience gained during the initial gap analysis and execution phase of the identified ageing management projects.

This procedure can be initiated by any new source of information on ageing that becomes available. This can be due to an internal event (like maintenance, inspection, malfunction, emergency or modification) or due to external events, which become known to the HOR. External events become known to the HOR by any of the IAEA databases on ageing (IAEA Research Reactor Ageing Database (RRADB) and IAEA Incident Reporting System for Research Reactors (IRSRR)) or by shared experience from a peer institute. Most of these experiences of peers are shared during the yearly meetings of operators of research reactors, like the AFR[29] and RROG[30] meetings. When a new source of

[29] Arbeitsgemeinschaft für Betriebs- und Sicherheitsfragen an Forschungsreaktoren
[30] Research Reactor Operators Group

information on ageing becomes available the relevance for the SSCs at the HOR should be determined and described in a quick scan report. When the quick scan determines that the newly identified ageing information is relevant to the HOR, the RCM analysis shall determine the approach for ageing management. The so-called phenomenon report of this RCM analysis shall not only describe the selected ageing management approach, but shall also describe how the identified degradation was interpreted and which mechanism is thought to be responsible. By having such a report, it is possible to deepen the understanding of the actual driver for the ageing mechanisms over longer periods of time. The format and content of both the quick scan report and phenomenon report is prescribed by dedicated instructions.

The outcome of the RCM analysis shall be approved by the head of reactor operations. When approving the selected approach, it shall also be determined how the new understanding is secured in the organisation by instructions or procedures. As with any instruction and procedure a revision period for these instructions and procedures has to be determined. When the time has come to revise the instruction or procedure the effectiveness of the instruction or procedure shall be evaluated and compared to the outcomes of the RCM analysis as captured in the phenomenon report. If there are striking discrepancies between the outcomes of the phenomenon report and the actual collected data over the revision period the phenomenon report shall be updated and the instruction or procedure adapted accordingly.

To facilitate the document control and track the revision periods of the individual instructions and procedures a new version of the document control system was implemented during the initiating phase of the ageing management programme.

Periodic improvement

The ageing management programme as described above was developed based on the outcome of the last 10-yearly evaluation of the HOR. Such a 10-yearly periodic safety review is already since 1996 a requirement for the HOR licence. More recently also the EU directive 2014/87/EURATOM demands for a periodic safety review which shall (amongst other items) include ageing. During the next 10 yearly evaluation the ageing management programme as a whole will be reviewed.

2.5 Licensee's experience of application of the overall AMP

2.5.1 Licensee's experience of application of the overall AMP - NPP Borssele

Particularly on the nuclear part of the plant the condition of the big structures and components like containment and the primary components is still very good after 44 years of operation and the current ageing management is focused on maintaining this.

The adequacy of the ageing management is confirmed by several internal (PSR and LTO assessment) and external reviews (IAEA: AMAT and SALTO).

However, supported by the reviews and experience feedback the organization realized that continuous improvement is necessary on ageing management.

Based on this, the ageing management at NPP Borssele is developing from a classic approach consisting of separate maintenance, surveillance and inspection programmes into an integral SSC oriented ageing management process in which coordination, traceability and experience feedback are very important.

Because of the German phase-out, NPP Borssele will seek international collaboration and information exchange more outside Germany in the upcoming years.

Engineers of NPP Borssele are quite intensively involved in international networks regarding LTO and ageing management and will continue to do this, both to support further improvement of ageing management at NPP Borssele and to support the ageing management at other NPPs in the world.

2.5.2 Licensee's experience of application of the overall AMP - HFR-specific information

In the first periodical safety review executed at HFR in 2003, the first ageing management related measures were defined. Main measures were the drafting of an ageing management plan and incorporation of ageing in the preventive maintenance procedures.

A large influence on the HFR asset management (and with that also the ageing management) were the events concerning (also see introduction in section 2.3.1.2):

- The deformations in the bottom plug liner in 2008;
- The tritium leakage in a buried pipe in 2012;
- The leaking gasket in the bottom plug seal in 2013.

After the last two events efforts on asset and ageing management were intensified to improve safety and reliability of the plant (e.g. implementation of an asset register, structurally more manpower in maintenance). In 2017 HFR is starting the development of a Long Term Operation Programme. The full implementation of this programme will take about 2 to 3 years. This programme will also undergo an assessment through a dedicated LTO mission of the IAEA (SALTO) in 2019/2020.

Current attention is drawn to continuing pool leakage (INSARR 2016 recommendation). NRG has started several actions such as automation of leakage monitoring, increased reporting and assessment of temporary leak rate changes, improved water-balancing of the pool and a pool liner inspection programme is being developed. These actions were followed by a visual inspection programme of the pool separation door guide frame (for which a PPO has been created) and concrete leak path investigations.

In 2017 NRG has done several further steps of the investigation: sampling of the concrete of the pipe corridor to look for water impact (carbonation). Conclusion was that overall the concrete was still of good quality. Concrete locations that are moist or have some damage have been and will be repaired. Further there are thoughts about investigating the corrosion of the aluminium from contact with concrete of rebar. In addition a CO_2 injection system for pool liner corrosion prevention has been installed. The system injects CO_2 in the void between pool liner and the concrete and hence will avoid corrosion.

2.5.3 Licensee's experience of application of the overall AMP - HOR-specific information

Despite the fact that the maintenance of the HOR reactor has continuously been evolving since the start of the reactor, a systematic approach to ageing management has only been developed in recent years. While developing this systematic approach to ageing management it became clear that a clear procedure for incorporating lessons-learnt (both from internal and external experiences) was not available. Such a procedure has been developed and the role of AMP coordinator has been formalised within the HOR operations department to make sure the ageing management procedure is executed.

Internal experience

In general, it can be concluded that the more robust a SSC is, the less focus it gets in a naturally evolving maintenance programme. This is only logical, since these robust SSCs do not require much maintenance. But when their safety function is critical (or long down times can be expected when such a robust SSC fails), it is worth investing the time to come up with an appropriate ageing management approach for such a SSC.

Furthermore, it became clear during the gathering of data on the ageing mechanisms of the individual SSCs, that a lot of information is available, but not easily accessible for trending or review. This is because a lot of information is collected and stored on paper. By setting up a digital database (as one of the projects of the initiating phase of the ageing management programme) for collecting and storing the data as collected during periodic inspections or tests, the gathering of information per SSC has become easier and trending can be done easily on the digital available data. This helps to determine acceptance criteria identify outliers in the collected data and to determine the effectiveness of the ageing management programme.

Existing procedures and instructions have been reviewed for their focus on ageing management and for providing clear acceptance criteria. When needed new procedures and instructions have been developed.

As quite a few new periodic/preventive maintenance actions have been determined by the ageing management programme a newly developed database for all periodic tests turns out to be essential to keep track of upcoming periodic tests.

External experience

Input and discussions on the way to shape, execute and maintain the ageing management programme based on the IAEA SSG-10 were held with some of the HOR peer nuclear research institutes. More specifically the operating teams of the BR2 reactor operated by SCK-CEN and the SAFARI reactor operated by NECSA were consulted.

Adequacy of the overall AMP

The HOR ageing management programme has been developed by the HOR as an outcome of the last 10-yearly safety evaluation (10EVA). Because of this the development of the programme and the progress was supervised by the ANVS. The development of the programme and the gap analysis was performed over the years 2012-2015 and agreement was reached between the HOR and the ANVS on the initial list of ageing management projects to fill in the identified gaps. The ageing management projects on this initial list have been completed by the end of 2016. By completing these initial projects the ageing management programme is considered to be implemented and operational. Since the AMP is just in operation for a short time, conclusions on its adequacy cannot be made yet. The authorisation of the ageing management procedure ensures that the ageing management programme is continuously improved. A first periodic review of the ageing management programme as such will be performed during the upcoming 10 yearly safety evaluation (10EVA around 2020 - 2021).

Some examples from the HOR operational experience with ageing

Some examples are:

- Radiation induced damage to the electronics leading to an offset of the primary flow sensor. The flow sensors were replaced and relocated to locations with low dose rate in 2004;
- Obsolesce of spare parts for the nuclear safety channels, this led to the complete renewal of the safety channels in 2010;
- Corrosion induced failure of a control valve in the primary system in 2016.

The last example has been described to some detail below.

The control valve is used to set the primary flow rate to its normal operating value. The valve was removed from the pipe for inspection and the reactor start-up was postponed. The valve is a butterfly valve with a passivated stainless steel blade, a stainless steel axle, an EPDM seat and a cast iron body. Due to corrosion of the body the axle had become stuck. Corrosion was found behind the EPDM seat inside the cast iron body. This specific type of valve is also used in several safety valves of the reactor; pool isolation valves in the primary cooling system and in the purification system. All valves had been installed in 1993 and 1994 and had been in operation for 22-23 years. It was decided to remove an easily accessible safety valve and inspect it for corrosion. Corrosion was indeed found in this valve too.

The valves have been replaced. The procedure for periodic testing has been improved and includes functional tests, a maximum lifetime for the valves is introduced and the AMP is adjusted accordingly.

2.6 Regulatory oversight process

Assessment of the overall AMP and modifications (a)

NPP Borssele

In the beginning of the 1990s, when the NPP was approaching 20 year of operation, a first PSR[31] was carried out. Also for the first time a report was requested on ageing management (AM). The modification of the licence to implement the PSR findings then also led to the introduction of a licence requirement on the overall AM in 1993. The regulator and licensee went through several iterations over several years to build the overall AMP. Main goal of the regulator was to stimulate the build-up according to international practice. At that time it was not common to have AM programmes, because it was thought that it was implicit part of the existing activities like maintenance, in service inspection, etc. Both regulator and licensee had to build up knowledge and experience. It was agreed that the licensee would have a systematic and working AMP before 1998.

In 2001 the regulatory body agreed to the plan and reference documents for the 2nd PSR (including using IAEA Safety Guide O10). In the beginning of 2003, about one year before delivery of the PSR

[31] Already before that date, in a backfitting project, major safety upgrades were made.

evaluation report, on request of the regulator an IAEA AMAT mission was invited. The goal was to have an international experience impulse to further improve the AMP. The findings were integrated in the PSR implementation plan (2004-2007). Around 2006 there was an agreement (covenant) between the Government and the licensee to allow for continued operation till 2034 (40 → 60 years) under the condition that always the current regulatory requirements have to be fulfilled. Although the licence has no time-limit, the FSAR was based on a design life of 40 years. Therefore the regulator requested from the licensee to apply for a modification of the licence based on new assessments till 60 years and a programme according to the IAEA SALTO Guidelines SR 57. This was supplemented with two Safety Factors of the PSR. The modified licence should be valid before 2014. Also it was agreed between the regulator and the licensee to invite IAEA SALTO missions. In 2008 IAEA, the regulator and the licensee concluded an agreement to start in 2009 with a first SALTO mission to assess the scope of the licensee programme. The scope had to be extended. In the period 2010-2012 the agreed documents of the SALTO programme, including a large number of component or system specific AMP's, were assessed by the regulator, supported by its TSO (GRS), the licence modification was applied for and in May 2012 a full scope SALTO mission was carried out. The licence granted, contained on one side a condition to implement the IAEA SALTO mission recommendations and suggestions before 2014. This was verified by a SALTO FU mission in February 2014. On the other side a number of other requirements related to LTO were included in the licence, covering actions to be completed before end of 2013 (40 y of operation) and actions to be completed before 2020. Also the assessments of related documents are done by the regulator supported by the TSO. In parallel also the OSART-missions in 2005 and 2014 payed attention to ageing management.

Research Reactors

The introduction and further stimulation of ageing management at the research reactors happened at a later moment than for the NPP Borssele. As in many other countries worldwide at that time, the regulatory attention for the safety of research reactors was relatively small. Also the development of IAEA standards for research reactors was lagging behind. On top of that for the regulator, at that time with a modest number of staff, in the Netherlands the focus first was on two large PSR projects for the NPP Borssele and the NPP Dodewaard. From the last part of the 1990s it was decided to introduce gradually, and first on a voluntary basis, the PSR and related to that AM at the research reactors. At first the focus was on the build-up of component/structure AMPs, more recently also work started on the overall AMPs based on IAEA SSG10.

At the HFR after the first PSR in 2005, a renewed licence was granted to the licensee NRG, containing amongst others conditions on PSR, AM (roadmap required) and 5-yearly INSARR missions (in which AM is a module). The first mission was in 2005. Subsequent missions were in 2011 and in 2016. Recently it has been decided to introduce INSARR Follow-up missions two years after the mission. The first one will therefore be in 2018. To further stimulate developments in AM, after Belgium, the Netherlands was the second country in 2015 to ask IAEA to develop a SALTO-mission for research reactors. It is the intent to have such a mission in 2019/2020 at the HFR, taking on board the experience of the first one in Belgium, which has been observed by ANVS in November 2017. ANVS will make sure that the findings of the Belgian mission will be input for both research reactors AMPs.

At the smallest research reactor HOR, the first PSR was finalized in 2002. An INSARR mission was carried out during the evaluation phase. Based on the second PSR-action plan an AMP based on IAEA

SSG10 was required, but this requirement is not yet transferred into the licence, because in the next years there will be a complete overhaul of the licence related to the next PSR around 2020. Since an INSARR mission was done only once and long ago it has been agreed to have another INSARR mission amongst the next PSR (around 2020) with attention for AM. It has also been decided to include a 10-yearly INSARR in the licence requirements. An LTO-programme and mission is not considered necessary for such a small reactor (graded approach).

Inspection of the implementation of the overall AMP (b)

For all nuclear installations there is an inspection plan. The inspection plan is based on the IAEA GS-G-1.3 'Regulatory Inspection of Nuclear Facilities'. Ageing management is one of the areas of inspection. It is applied according to a graded approach. For the coming 3-4 years at the NPP part of the inspections is at least on the licence conditions for LTO. On the other hand there is a need to develop a more detailed AMP multi-year inspection programme. For the research reactors the focus will be on the completion of the integrated state-of-the-art AMP of the HFR, because it will still take a couple of years, and on the AM-related findings from the INSARR mission. Based on a recent inspection, the ANVS will require the HFR licensee to present a detailed implementation plan, with milestones and periodic reporting. Since the HOR AMP was accepted by ANVS in 2016, like for the NPP, now a more detailed inspection plan needs to be developed.

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

2.7.1 Regulator's assessment of the overall ageing management programme and conclusions - Matters common to all installations

The three described installations all are in different maturity phases of their AMPs. Where the NPP is complying with the state of the art, the HOR has just completed its implementation in 2016 and HFR will need years to complete this, including the ambitious LTO-part. Because of this, the crucial points of the AMP can be determined. Initially, it is important that the framework of the AMP is in place and allows for enough flexibility. This is required because it is likely that not all ageing mechanisms are found at the start of the AMP. It should allow adding of newly found ageing mechanisms or SSCs.

A second important common topic in the process of establishing an effective AMP is the embedding in the organization. For existing installations that incorporate an AMP during their lifetime, it is observed that it is a challenge to embed ageing management in the integrated management system. The challenge is to merge ageing management with the existing maintenance part of the organization.

Overall, the ANVS is satisfied with the AMPs developed (NPP, HOR) and the one being in the last stage of development to an integrated AMP (HFR). ANVS considers the HFR goal to be ready by the SCO-mission as ambitious and therefore will watch closely the progress and sufficiency of manpower made available. A positive signal is that no reactive inspections have taken place in the last few years. However, the AMP is a process, not a project. Because of this, continuous improvement is required and that is reflected in the policy and inspection plan of the ANVS. Main efforts of ANVS have to focus on the finalization of the LTO-actions of the NPP, the monitoring of the HFR progress and developing multi-year AM inspection programmes.

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 Scope of ageing management for electrical cables - NPP Borssele

Introduction- description of ageing management programmes for electrical cables at the NPP

In conformity with the methodology of the ageing management process a leading document on AM of cables is drawn up, the so called ageing description. This document can be seen as the top level document. In this document there is information about the scope, the used components, process- and environmental conditions, ageing mechanisms & degradation effects, ageing management methods, strategy on ageing management, executed measures in the past (history), an evaluation regarding the nine generic attributes of an effective ageing management programme[32] which includes the actual AM activities, as well as the responsibilities for the different departments regarding ageing management.

Periodic repeated activities serving ageing management are part of the preventive maintenance programme.

The combination of the ageing description and the relevant periodic activities can be regarded as the ageing management programme for cables and wires. In this context a wire is defined as a single conductor with singular insulation. A cable is defined as one or more insulated conductors with a protection sheet.

The ageing management programme as defined above focusses on the existing wires and cables in the plant. Nevertheless proper ageing management starts with the choice of equipment which is qualified for the stated requirements and for the time needed. Plant design rules, e.g. redundancy, assure the process control (safety functions), and by that nuclear safety, in case of a single failure.

Actual scope of ageing management for electrical cables at NPP Borssele

The main goal of ageing management for passive electrical commodity groups is to maintain the level of nuclear safety. Ageing management should ensure a sufficient degree of availability of the SCs important to safety and in particular to avoid any significant increase of ageing related common cause failure potential. Consequently all the cables important to safety are part of the scope.

All the systems and components at NPP Borssele[33] (KCB) are classified in line with their importance for nuclear safety (safety, safety relevant or non-safety). Within the scope of ageing management are those cables and wires which are relevant for the classified functions of components important to nuclear safety (e.g. closing a valve, starting a pump, etc.).

The ageing management of cables and wires is described in the document "*Verouderingsbeheersing elektrische draden en kabels*". Electrical cables and wires are one of the passive electrical commodity groups as defined in the Long Term Operation Assessment (LTOB) screening process. To manage the

[32] Ageing Management for Nuclear Power Plants, IAEA Safety Standards Safety Guide NS-G-2.12, 2009
[33] In Dutch known as Kerncentrale Borssele, acronym KCB.

effects of ageing degradation it is not necessary and not efficient to focus on each individual cable. Groups of cables or wires with, for example, corresponding materials, age and environmental conditions can be treated as groups.

To support the approach of ageing management the cables and wires are divided in component groups and subgroups.

Based on voltage level:

- Low voltage: ≤ 1000 VAC (between phases) or ≤ 1500 VDC (between the poles);
- Medium voltage: > 1000 VAC and ≤ 30000 VAC (between phases);
- High voltage: > 30000 VAC.

Based on application:

- Low voltage wiring
    - Spreader wiring;
    - Wiring in electrical cabinets;
    - Internal wiring in (active) components.
- Low voltage cables
    - Power cables;
    - Signal cables.
- Cables for specific applications, e.g.:
    - Cables with design base accident requirements;
    - Coax cables like used in neutron flux instrumentation.
- Medium voltage cables.

Note: the high voltage transmission lines which connect the plant to the local grid are not part of the ageing management programme for cables and wires. Inspection and maintenance on the high voltage transmission lines is performed by the electrical grid operator Tennet.

3.1.1.2 Scope of ageing management for electrical cables - HFR-specific information

Based on the methodology described at chapter 2 (HFR-specific sections 2.3.1.2, 2.3.2.2, 2.3.3.2, 2.4.2 and 2.5.2) a review of the relevant electrical systems has been made. The list below summarizes the scope related to electrical cables which have been reviewed. The relevant ageing mechanisms and analyses are described in section 3.1.2.2. Cable test procedures are described in section 3.1.3.2. Preventive actions, control measures and recommendations are described in section 3.1.4.2.

The AMR scope covered cables, wires and connections of electrical and instrumentation systems.

A wire consists of a single electrical conductor covered with insulation. The main application of the wire is the interconnection of equipment inside electric or electronic cabinets or as cross wiring in marshalling racks.

A cable consists of a single conductor or multiple conductors with their own insulation, with cable filler between the conductor insulation and this assembly surrounded by a cable jacket in order to provide environment protection over the inner assembly.

A connector is a coupling device attached to two or more wires or cables for the purpose of connecting electric circuits without the use of permanents splices.

AMR cable scope:

- 10 kV cables
- 380V/ 220V cables
- Instrument and control cables (≤ 110 V DC)
- Nuclear instrumentation cables
- Electrical (distribution and connection) cabinet wiring (≤ 1 kV)
- Cross connection wiring (≤ 110 V)

Cable details are listed in the next table.

Table 3 Cable types and details, HFR

No	Cable type	Use	Subject to radiation
1	All weather Super CAT (data instrument cable)	High range monitor	None
2	Jobarcoflex cc	Several	None
3	Draka DC1*Dataflex yy* (data transmission)	Several	None
4	ELDRA Signal cable JY(st)Ymb	Several	None
5	H05VV5-F DRAKA 05 2006 241763 SFmb	Power supply cooling water pumps	Radiation load during operation
6	Draka YMvK (1963 PANEL 2 CR near magnet power supply)	Several	None
7	Draka signal cable mb 2906	Several	None
8	Draka VMvK (1968 48V= Power supply RBG)	Several	None
9	ACEC Charleroi VMvK (1963 Reactor hall EXP. Measure system)	Several	None
10	Eupen VMvK (1963 Reactor hall EXP. Measure system)	Several	None
11	DAETWYLER Pyrofil Keram JE-H(ST)H..BDFE180 E30-E90	Control rod regulation	Radiation load during operation
12	Draka cable Vultmb	Several power supply	Radiation

No	Cable type	Use	Subject to radiation
		systems in radiation area’s	load during operation
13	Draka cable H05VV5-F SFmb (2007)	Control circuit systems in radiation area’s	Radiation load during operation
14	Draka cable SFmb	Control circuit systems in radiation area’s	Radiation load during operation
15	Draka cable QWPK/H07BQ-F 450V/750V	Several	None
16	DRACODAmb (DRAKA cable)	Several	None
17	DRACODAmb (DRAKA cable)	Several	None
18	Draka 08 instrument cable mb 9104	Wiring of several instrumentation systems partly located in radiation area’s	Radiation load during operation
19	Jobarcoflex CKY ce	Several	None
20	Draka 03 NWPK special/H07RN-F 450V/750V	Power supply of electrical motors	Radiation load during operation
21	Draka 05 147865 DATAX-FLEX LiYCY 300V/500V	Serval	None
22	Draka DCI DATAX-FLEX YY (data transmission cable)	Instrumentation wiring	None
23	Draka DCI DATAX-FLEX YGY (data transmission cable)	Instrumentation wiring	None
24	Draka 03 EDRATEEN pipe line 110ºC/ H05GG-F 300V/500V	Several	None
25	COAX RG62 A/U MIL-C-17	Nuclear instrumentation cables in the reactor pool	Continuous radiation load including neutron radiation
26	Draka cable H07V-K (single wire)	Serval	None
27	UTP - UC300 24 cat 5e u/utp 4p IEC 61156-5 LSHF	Several network connections including radiation area’s	Radiation load during operation

No	Cable type	Use	Subject to radiation
28	DRACODAmb (DRAKA cable) 2704	Firefighting alarm system located in radiation area's	Radiation load during operation

3.1.1.3 Scope of ageing management for electrical cables - HOR-specific information

One of the conclusions when implementing the Ageing Management Programme as described before was a lack of ageing management for cables. Therefore, a cable ageing management strategy has been developed since 2012. This cable ageing management strategy is based on the IAEA guide Nuclear Energy Series, No. NP-T-3.6 "*Assessing and Managing Cable Ageing in Nuclear Power Plants*". A graded approach to this guide was considered for the HOR research reactor. Grading with respect to the requirements of the above IAEA guide for nuclear power plants becomes possible at the HOR because the analysis to come to the selection of a representative set of cables shows that the HOR is not dependent on electricity for the three main safety functions (safe shut down, cooling and isolation). Therefore qualifying cabling against a Design Base Accident is no requirement.

Due to the fact that every nuclear facility utilizes a vast amount of cables it is unpractical and unnecessary to asses each cable individually in an aging management programme. In order to select a representative set of cables for the HOR cable ageing management programme, the IAEA guideline is being used.

The IAEA guideline states that individual cables should not be monitored but grouped and focus should be given to the safety relevant cables. The WENRA technical Specification document for the National Assessment Report also urges to focus on the safety relevant cables. The IAEA guide is however more elaborate in the description of the method and criteria used for selecting the cables within the scope of ageing management.

This selection of a representative set of cables is based on the grouping of the function of the related equipment and on the environment the cable has to go through. The purpose of such environmental qualification is to demonstrate the equipment (and associated cables) installed in an area can perform their expected (safety) functions throughout their life. This type of grouping of equipment based on the required safety function in different environments is used in more IAEA guides.

The categorization in the IAEA guide deviates from principle for cable segregation during the design of the HOR. The HOR cable segregation is based on the regular safety class classification, as determined for the HOR. In general, the HOR safety classes are utilized in order to guarantee the three main safety functions for safe shut down, cooling and isolation. The HOR is designed in such a way that none of these safety functions require electricity. Furthermore cables are installed fail-safe. If a signal cannot be executed fail safe, such a signal is executed redundant or diverse and the cables carrying such signals are segregated. Therefore the guaranteed functioning of a cable is never a requirement for safely shutting down, cooling the reactor after shutdown or isolation the reactor environment from the ambient environment in case of a possible release of radioactive material.

If the HOR safety classification system would be used than all the cabling would be interpreted as non-safety critical since they are not paramount for the actuation of the three main safety functions. To still be able to make an informed selection of cables the IAEA guideline is therefore used, instead of the HOR safety classification system.

Selection of a relevant set of cables

Following the IAEA guideline, the first task is to group the HOR equipment into equipment categories. The cables associated with the equipment which require electricity are than categorized accordingly. The requirements of the equipment (and cables thereof) can be categorized by their function and expected operating environment; which includes both regular as accident situations.

Equipment categories

The following equipment categories are defined in the IAEA guideline for the qualification of electrical equipment:

Category 1: Equipment relied on for mitigating the effects of an accident (safety systems) that are subjected to the accident environment.
The equipment and associated cables that are relied on for the mitigation of an accident and are also subjected to the accident environment have to demonstrate that they are capable of operating during and after the worst case Design Base Accident (DBA).

Category 2: Equipment relied on for mitigating the effects of an accident (safety systems) that are not subjected to the accident environment.
This equipment and associated cables have relatively lesser performance requirements because they remain accessible for servicing subsequent to the DBA.

Category 3: Equipment necessary to prevent the release of radiation.
The third category of equipment and associated cables is generally only required to complete their active function.

Category 4: Equipment that supports the systems necessary for accident mitigation.
The fourth category of equipment and associated cables provides supporting functions necessary for continued operation such as air supply, lubrication systems and service water for cooling, electric and hydraulic power.

Category 5: Post-accident monitoring instruments.
The fifth category of equipment and associated cables supports monitoring and gauging the potential for a containment breach or off-site release and provides necessary information for accident management and evacuation requirements.

Category 6: Equipment necessary for the normal operation of the plant.
The sixth category of equipment and associated cables is necessary for supporting power production, avoiding plant trips or transients.

Category 7: Equipment that can fail and mislead the operator during an accident.
The seventh category of cables and equipment needs to remain functional to prevent the provision of misleading information to the operator.

Operating environment

After grouping the equipment (and thus the associated cabling) into these equipment categories, the cables will be classified according to their operating environment. When classifying the cables according to their operating environment it should be considered if the cable should remain its function in an accident environment. Environments can be characterized by mild or harsh. But this is a relative qualification, as mild is the accepted environment during normal operation. Of course, a cable needs to be selected according to the environment in which is supposed to operate. A harsh environment can be created in a postulated accident scenario, due to increased levels of radiation, temperature or exposure to water or chemicals.

From the combined set of equipment categories and operating environment the most conservative selection is made to form a representative set of cables for ageing management. At least all the applied cables with a safety function will be included in the cable ageing management programme.

The results of the above described selection procedure for the situation at the HOR can be found below.

HOR specific cable information and selection

Before making the cable selection based on the above described method, some HOR specific cable information shall be provided in this section.

External power is supplied to the Reactor Institute Delft (RID) by 10kV voltages lines, which are stepped down to 380VAC with 3 transformers on the RID premises. From there the 380VAC is fed to the low voltage supply room from where the electricity is distributed to the rest of the Reactor Institute. In case of a power grid failure an on-site diesel generator is able to supply continuous power. Cabling is running from the generator to the low-voltage supply room.

Despite the fact that the HOR reactor is in operation since 1963, very little cables are still in use from that date. The main replacement in cabling was done when the control room was moved from within the reactor containment to outside the reactor containment in the years 1980-1982. Besides moving the control room, the reactor safety and control system was updated including the reactor instrumentation. Nearly all instrumentation, control cabinets and associated cabling was replaced during this update. Over the last couple of years, the low voltage supply room, the electrical cabinets within the reactor containment and the interconnecting electrical cabling between the electrical cabinets in the reactor containment and the low voltage supply room have been replaced.

As most of the cables still in use were installed when the control room was moved and the instrumentation updated, the strategy for segregating cabling based on their function, during that project will be explained here.

Cable segregation is done for the following groups of cabling:

- GI – General Instrument,
- GC – General Control,
- 380/220VAC,
- 110VAC,
- ASC – Safety Control “A”

- BSC – Safety Control “B”
- ASI – Safety Instrument “A”

The applied cable type is selected based on the above segregation in functional groups. Of course, deviations to the applied cable type will be made if the application of the cable or environment requires so. Examples of special applications are high voltage cables or high frequency cables. In the case of the HOR both of these exceptional applications can for example be related to ionisation and/or fission chambers.

A special environment is primarily an increased irradiation field, as high temperatures and or high pressures are not present in case of an open pool type reactor, like the HOR. The application of cables within a flexible cable conduit (like required for cables running to the moving reactor bridge) is an example of a special environment for the HOR.

Not considering the above special applications or environments the following overview of applied cables can be given for the above mentioned segregations.

Table 4 Segregation and cable application at the HOR

Segregation	Multi core cable	Flexible cable conduit
GI	DEF STAN C	DEF STAN C
GC	318	DEF STAN C
380/220VAC	694	DEF STAN C
110VAC	DEF STAN A	
ASC	694	DEF STAN C
BSC	694	DEF STAN C
ASI	694	DEF STAN C

DEF STAN 61-12-part 5 cable

Cables made according to the DEFence STANdards 61-12-part 5 as used at the HOR are manufactured by the British Insulated Callender’s Cables (BICC) company. DEF STAN A cables have so called Low Tension (LT) unscreened cores, laid-up and are sheated. DEF STAN C cables are LT unscreened cores, laid-up collectively screened and sheated. In the DEF STAN code, it can be found that LT cores are insulated with PVC and are suitable for operation up to 440V rms at frequencies up to 1.6kHz. Also the sheath is made of PVC. The type of PVC as is used in the DEF STAN is determined by the British Standard BS-6746.

UK CMA reference coded cable

In the UK references a cable is identified by a combination of 4 (or sometimes 5 numbers), which can be followed by the addition of some letters. In the above table only the first three digits are stated as the last digit only indicates the number of cores. The first digit in our case can be 3 or 6 indicating a maximum voltage application of 300/500V or 600/1000V respectively. The next two digits are either

18 or 94 in case of the HOR. 18 indicating an isolated and sheathed round cable and 94 an isolated sheathed, single wire armoured, sheathed round cable. In the cable lists both the addition of a letter “X” and “Y” can be found, both indicating a PVC isolation and “Y” also ensuring a PVC sheath. All applied UK reference code cables are fire retardant according British Standard BS-4066. Most of these cables have a 1,5mm^2 conductor made of 30 strands of 0,25mm. The UK CMA reference only provides information on the general characteristics of the cable. In the “*Contract for the new safety and control instrumentation for the Hoger Onderwijs Reactor*” it can be found that cables are actually manufactured according to British Standard BS 6346.

From Table 4 it also becomes clear that no higher voltage than 380VAC is required as a supply voltage to operate the HOR.

Relation to other loss of power analyses

As mentioned before the selection of a representative set of cables is based on the IAEA NP-T-3.6 guide. The analysis that is carried out for the categorization of the equipment and associated cabling is in large part similar to the Loss Of Offsite Power (LOOP) and Station BlackOut (SBO) scenarios, as performed in the Complementary Safety Margin Assessment of the HOR (better known as the Fukushima stress test or CSA). In determining the equipment category for the various SSCs the reasoning and conclusions of the CSA are therefore used, as for the functioning of the equipment it does not matter if the equipment fails due to a power loss or due to a failing cable.

Equipment classification at the HOR

- Category 1 electrical equipment
  The design of the HOR is such that the three main safety functions are performed without the need for any electricity supply or electrical backup system. This conclusion is endorsed by the national report HOR on the Complementary Safety margin Assessment for the combined scenarios of LOOP/SBO and Loss of Ultimate Heat Sink (LUHS).
  None of the HOR cables is therefore included in this category.
- Category 2 electrical equipment
  What is stated above for electrical equipment in an accident environment, also applies for this category which does not include the accident environment.
  None of the HOR cables is therefore included in this category.
- Category 3 electrical equipment
  As already concluded in category 1 the isolation function is performed without the need for electricity. In the case of isolating the containment, the isolating action is performed by the energy stored within the system (spring or air buffers). Also, the other three safety barriers (fuel matrix, fuel cladding and pool), as mentioned in the national report HOR on the Complementary Safety margins Assessment are not dependent on electricity.
  No electrical equipment or related cabling is therefor included in this category.
- Category 4 electrical equipment
  Despite the fact that none of the equipment in the above categories is dependent on electricity at the HOR, electricity is of course needed to make informed decisions in both normal and accident scenarios. In case of a power grid failure an on-site diesel generator is able to supply continuous power. The cables running from the generator to the low-voltage supply room and

subsequently to the electrical cabinets which are located in the reactor containment, are therefore eligible for cable management in this category.
In case air pressure is needed for accident mitigation (for example for the opening of isolation valves to introduce water to the pool) dedicated air buffer tanks are installed locally. Active air supply, lubrication systems or hydraulic power are not relied upon for accident mitigation.
Cooling water in case of severe leakage is taken from the storage tank (which is located elevated compared to the basin) and can contain as much water as about 85% of the reactor pool. The storage tank can be refilled using the municipal water supply.

- Category 5 electrical equipment
  Post-accident monitoring is currently being done by the same equipment, which monitors the containment during normal operation. The containment pressure, temperature and area dose rate readings are monitored in the main control room. But the measuring ranges are not calibrated for possible accident scenarios.
  Pressure built up till levels beyond the design specifications of the containment is actually not a possible scenario as the containment is protected against both under pressure and over pressure by a water lock, as a last line of defence.
  For this category, the cabling associated with the radiation dose rate monitor located under the reactor bridge is selected as a conservative NAR example or representative of this group of electrical equipment, due to its function and high voltage supply cables and low amperage, high count rate signal cables. In this case the accident environment can be classified as harsh in the case of an accident due to the direct line of sight to the most likely source of radiation in case of a severe accident; the reactor core. This is of course when the reactor is no longer covered by pool water.
- Category 6 electrical equipment
  Equipment needed for normal operation is determined by the minimum requirements for operating the HOR reactor. Such requirements can be found in the operating permit and the technical specification documents related to the permit. Both control/monitoring systems and reactor protection systems are described in these documents. As a conservative representative of this category the so-called safety channels are chosen which monitor neutron flux levels relatively close to the core by using boron coated ionisation chambers. These safety channels are part of the reactor protection system. This representative is conservative as the instruments operate in a raised radiation field, have to perform their task underwater and the supply cabling is high voltage while the signalling cabling is low amperage and high frequency. As this category implies this equipment plays no role in case of an accident scenario, the environment can even though be judged as harsh.
- Category 7 electrical equipment
  This category is for electrical equipment that can fool or mislead the operator. Also within this category a lot of instrumentation can be included. As a conservative representative of this category the fission chamber is chosen. This channel is relied upon mainly during start up or when operating the reactor in natural convection mode. As with the safety channels the operating environment of the equipment involves raised radiation level and a high moisture levels as the detector is being used underwater.

3.1.2 Ageing assessment of electrical cables

3.1.2.1 Ageing assessment of electrical cables - NPP Borssele

General

AM of the passive commodity group “Elektrische draden en kabels” is primarily focused on the electrical insulation. The electrical insulation is the essential factor in the functionality of wires and cables. Ageing degradation of insulation may lead to a short circuit and by that to unavailability of components. Also can a decrease of insulation resistance lead to an increase of leakage current. Particular for measurements with low current signals this can influence the accuracy of measurements in a negative way.

Mechanical protection is the mean function of cable jackets and armouring. As long as the jacket is not damaged, ageing degradation is not considered as highly essential for the function of the cable. Also ageing degradation of the copper or aluminium conductors is not regarded as significant for the functional behaviour during the plant life including Long Term Operation (LTO).

For the functionality of the electrical chains it is essential that the cable terminations and connection to clamps remain in good condition. This is a subject in AM of the commodity group “Elektrische verbindingen”, in which conductivity is a point of interest.

In the next part of this chapter the ageing assessment and the measures to manage ageing are described for the different groups of cables.

3.1.2.1.1 Qualified life assessment for cables with a functional requirement during design base accidents

A specific group of components are those within the scope of Design Base Accident resistant Electrical Equipment (EQDBA), which have a requirement regarding functionality during a postulated Loss of Coolant Accident (LOCA) or High Energy Line Break (HELB). For each component the whole electrical component chain, which includes cables, must be functional accordingly to the requirements. These requirements includes the harsh environment conditions and a specific time period for which the components must be functional. In the LTOB project this subject was treated as Time Limited Ageing Analysis (TLAA).

The cables in scope are an important group in relation to ageing degradation. That is because ageing degradation could lead to common cause failure during harsh environment conditions, whereas the functionality during normal operation is not threatened. The demonstrable assurance of functionality of these cables requires a programme for each single cable in the scope. AM is involved in this programme, Figure 12 demonstrates the process.

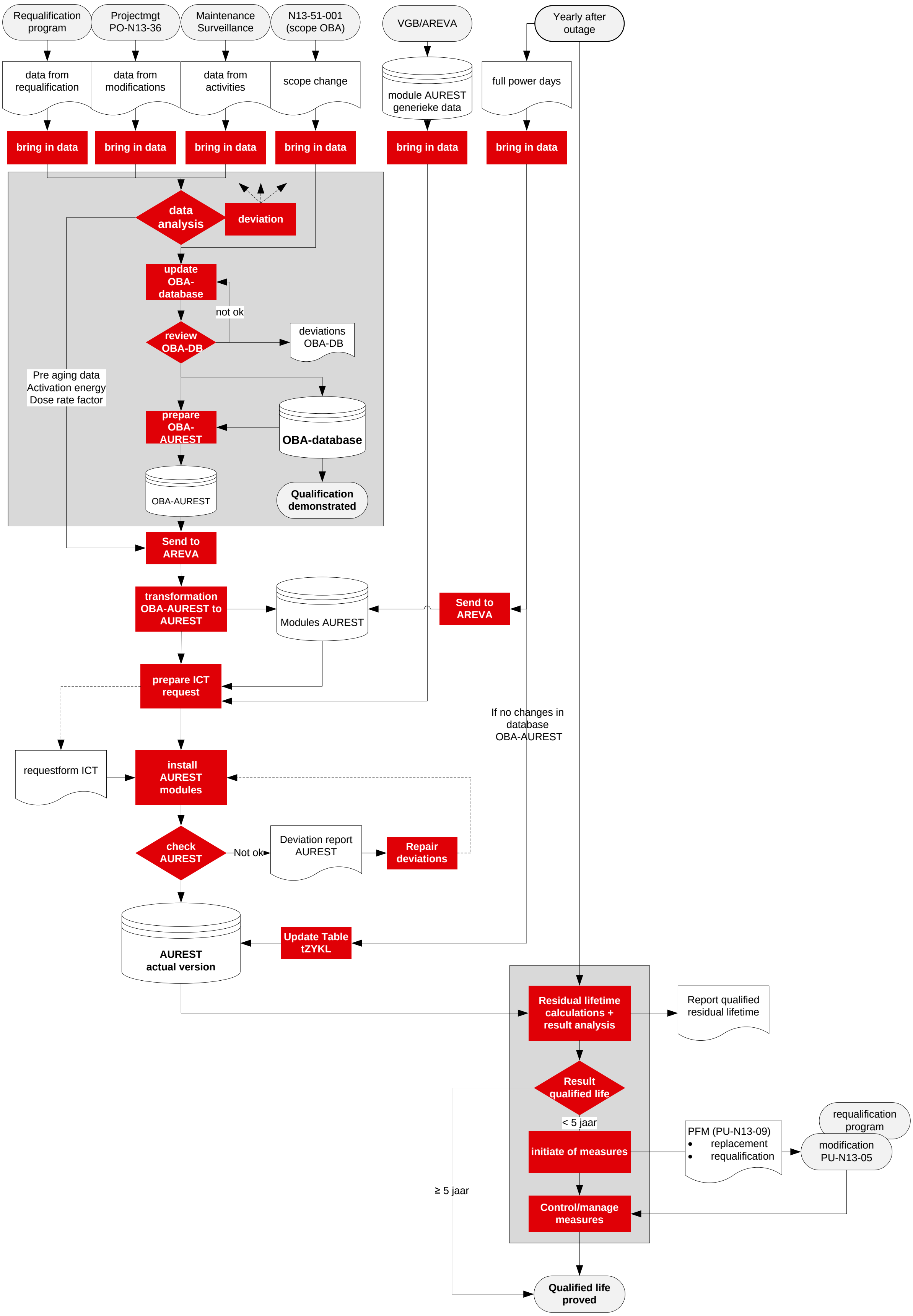

Figure 12 Flowchart of EQDBA process at NPP Borssele

Environmental qualification starts with the installation of type tested equipment in conformity with appropriate and accepted standards, e.g. KTA or IEEE. Artificial ageing of components is part of the type test. The data of the artificial ageing programme is used to calculate the qualified life of the components at their location in the plant.

On this subject KCB participates in the international working group (WG) „Betriebsbegleitende Nachweise der KMV-Störfallfestigkeit“ (BBNKMV), which is in the organization of VGB Powertech. In this WG several Nuclear Power Plants (NPPs) cooperate in a programme to bring evidence of qualified life of components according to the German standard KTA3706. In the next years KCB will anticipate on the German nuclear phase-out, which also includes the termination of the working group. Cooperation with other Siemens-KWU built plants will be obvious.

To calculate the qualified life the tool AUtomated Residual lifetime ESTimation (AUREST) is in use. In cooperation with the working group BBNKMV, Areva NP developed and maintains the AUREST tool. Calculation of the thermal qualified life in AUREST is based on the Arrhenius equation. The radiological qualified life is based on the equal dose - equal damage principle, with respect to the dose rate effect if it is relevant for the used materials. Figure 13 gives an example of the result of the AUREST analysis and calculations, in this case of the component chain of a steam generator level measurement.

KCB Restlebensdauern radiologisch und thermisch Die Berechnung erfolgte bis einschliesslich des Zyklus 2016

Funktionskette: YB001L002 FK Nr.: 287 NIVOMETING PLM SL- RLX Druck- und Differenzdruckmessumformer

Nr.	KKS Anz. Volllasttage Qualifizierung	Hersteller Prüfbescheinigung radiologisch Prüfbescheinigung thermisch	Kategorie	Typ Prüfergebnisse radiologisch Prüfergebnisse thermisch	Raum DL-Einbauort [Gy/h] Temp.-Einbauort [°C]	Einbaujahr Quali-DL [Gy/h] Quali-Temp. [°C]	Akt.-energie [eV] Quali-Dosis [Gy] Quali-Dauer [h]	Exp.-DL [-] LV rad. [a] LV therm. [a]	Qual >= Anf RL rad. [a] RL therm. [a]
10	YB001L002	Rosemount	GEBER	1154-HP5-RBN0037	01.213	1997	0.00		OK
	6.264			TÜV FRW116696Q,KWU 37.07PB	1.57e-04	1.12e04	5.50e05		>60
	PLL	pauschal 40 Jahre inclusief 20% marge			24	95	420,480		18.40
20	YB001L002	Qualtech	Stecker	QDC	01.213	2013	0.90		OK
	941			PEI-TR-880701-04	1.57e-04	1.00e04	2.00e06		>60
	PLL			PEI-TR-880701-04	24	263	1		>60
30	1KX28025	Rockbestos	Kabel	2/C 20 AWG 600V Shielded Firewall II		2013	1.34		OK
	941			QR-5805	1.57e-04	4.00e03	2.00e06		>60
	PLL			QR-5805	28	121	168		>60
40	1KX001F002	Weidmüller	Reihenklemme	KMV-F niedrige Form	01.324	2013	0.97		OK
	941	FIL-ETL1-07-0016-a u. FIL-ETL1-09-0025-a		NGLE/2002/de/0011	1.57e-04	1.10e02	8.00e04		>60
	PLL	FIL-ETL1-07-0016-a u. FIL-ETL1-09-0025-a		NGLE/2002/de/0011	28	115	648		>60
50	1KX8101	HEW	Kabel	JE-LI2GC2G ..FRNCX(s) / JE-2G(St)2		1986	1.11	0.20	OK
	9.551	IS-ETL1-MUC/may-grIS-ETL1-MUC/may-gr		NLEC-G/2006/de/0032d	1.57e-04	6.70e-01	6.10e04		>60
	PLL	116-584-ELA-84-3a		Q/8/7730-04-0-03	28	135	168		>60
60	XG012G234	Schott	Durchführung	T 61	01.324	1986	1.25		OK
	9.551	ETL 10/PB 308/92 u. ETL 10/PB 302/92		Protokoll 8403-01-223	1.57e-04	5.00e02	5.00e04		>60
	PLL	ETL 10/PB 308/92		Protokoll 8403-07-214	27	130	360		>60
70	1LN8001	s. Prüfbericht T12-08-ETI	Kabel	JE-Y(St)Y...		1972	1.25		OK
		T12-08-ETL004		NLEC-G/2007/de/0028 u. T12-08-ETL004	2.53e-04				
	RLI	T12-08-ETL004		NGLE/2004/de/0012a	28	85	2,401		>60

Hinweis: Funktionskettenglied kann nicht berechnet werden. Siehe Protokoll

AREVA NP / PTCQ-G | Funktionskettentool: FK2012a, Anlagenübergreifende Gerätebibliothek: GB2016a, Anlagenspezifische Gerätebibliothek: GB2016b | Die Berechnung erfolgte am 5-1-2017 16:13:21, Angewandter Sicherheitsabschlag: 20% | Page 310 of 374

Figure 13 **Example of remaining qualified life calculations with AUREST at NPP Borssele**

Specific for cables the ageing data is verified and periodically updated in an on-going qualification process with use of a cable deposit, which is located at the main coolant line in a German PWR NPP. It is to decide how to handle the future of the cable deposit after the nuclear phase-out of the German NPPs.

Due to the analysis in the light of Long Term Operation (LTO) of the KCB a replacement programme of Design Base Accident resistant Electrical Equipment was carried out. For example cables of the main coolant temperature sensors are replaced while the radiological qualified life wasn't sufficient for the period of long term operation until 2034.

In the outage of 2017 a cable of the reactor pressure vessel level measurement is replaced due to the ending of the thermal qualified life, determined as a measure in the continuing process.

3.1.2.1.2 Ageing assessment of cables and wires

Ageing includes processes and mechanisms, which lead to the degradation of materials and by that changes in the original, designed or as-built characteristics of components, e.g. cables. The degradation of materials is characterized by changing material properties, which may be relevant for the functionality.

In contrast to the cables with a LOCA requirement the probability of simultaneous failure of cables due to ageing degradation is very low. As a result the impact on nuclear safety of ageing degradation of non-LOCA cable is much smaller. The design of the plant captures the effects of single failure of an electrical circuit.

In principle the operational time of the plant, and of the SSCs, was planned to be 40 years. For the period of LTO an assessment is carried out to determine the reliability of safety and safety relevant SSCs, including cables. While it was found that it is for the most of the non-LOCA cables difficult to determine a reliable operational lifetime based on their specifications, in the analysis for LTO a material based approach was developed for the period of LTO.

In conformity with the procedure in the conceptual document[34] (Figure 14) the 10 steps described below were carried out.

[34] Conceptual document LTO "Bewijsvoering" KCB, NRG-report NRG-22701/10.103460, 9 September 2011

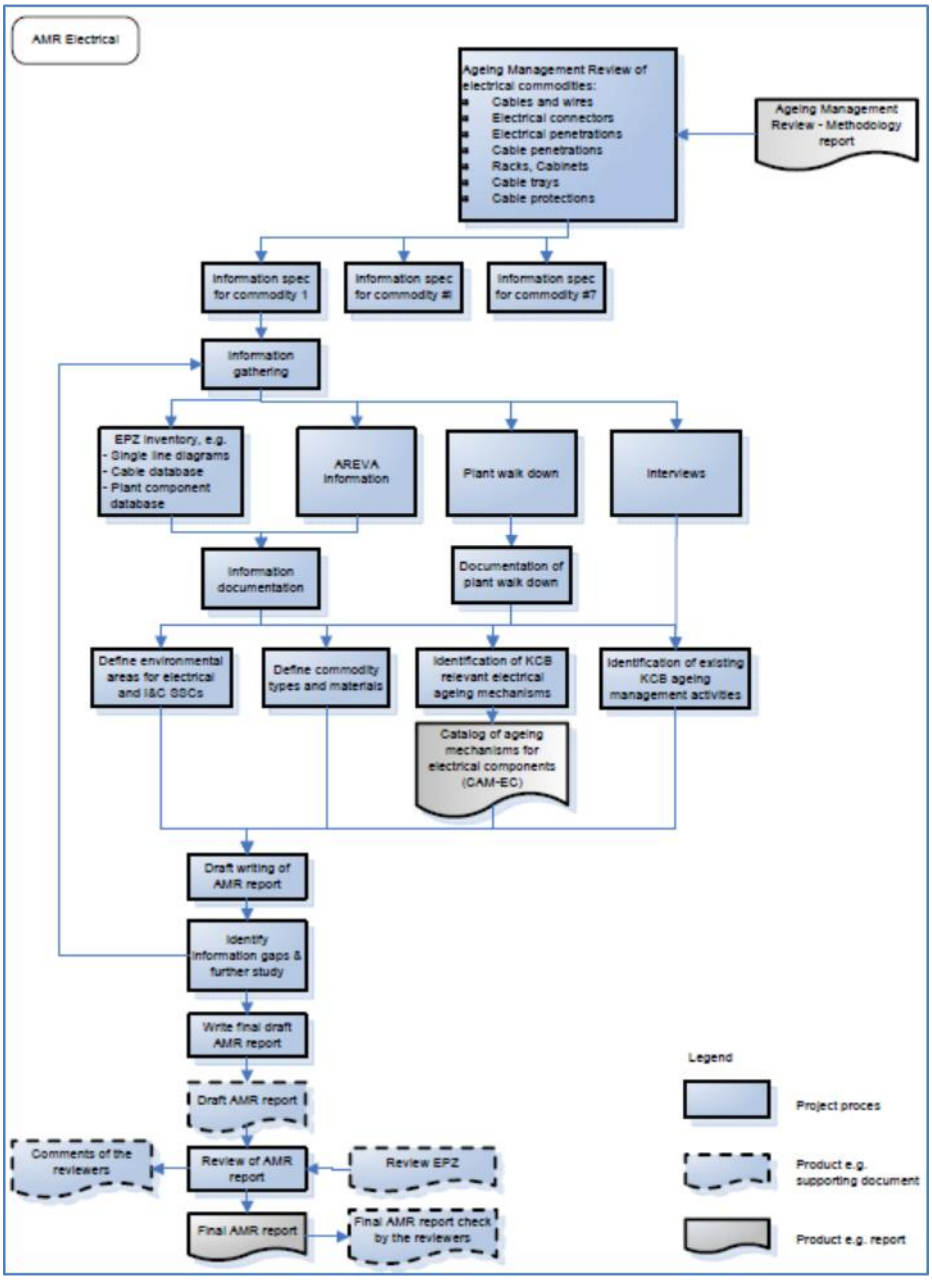

Figure 14 LTOB-AMR electrical assessment at NPP Borssele

Step 1: Identification of Component Materials

All the wire and cable types which are in use at KCB are listed up and are investigated for the applied materials for insulation, jacket and conductor, Figure 15 gives an example of an overview of a wire and cables which are polyethylene insulated.

Cables - Low-Voltage:
PE (Polyethylene)

Insulation Material	*Jacket Material*	*Conductor Material*	*Type*	*Application*
PE		Cu	PEI/PSCR/OSCR/SWA	LV
PE	EVA	Cu	FRNCX-JE-LiHCH-LgSi	LV
PE	EVA	Cu	JE-LiHCH	LV
PE	PE	-	HDPE 40MM2 GRIJS	Rohr
PE	PE	-	HDPE 40MM2 ROOD	Schutzrohr
PE	PVC	Cu	PTT-NORM 88	
PE	PVC	Cu	BELDEN 8760	LV
PE	PVC	Cu	COAX RG58	LV
PE	PVC	Cu	COAX RG59	LV
PE	PVC	Cu	COAX RG59 b/u	LV
PE	PVC	Cu	PTT-NORM 88	LV
PE	PVC	Cu	RE-2Y(St)Y-fl	LV
PE	PVC	Cu	VO-YMvKB-as	LV
PE	PVC	Cu	YMeK-rv-as	LV

Figure 15 Example of wire and cable material identification at NPP Borssele

The investigation resulted in the following overview of relevant materials and applications of the passive commodity group wires and cables:

Commodity Group	Application	1. Material	2. Material	3. Material
Cables	Low-Voltage Cables • Power-Supply • Signal • Thermo-Measurement	Insulation Materials: NR, PE, XLPE, PP, PVC, PTFE, FEP, SIR, PI, TPC	Jacket Materials: NR, CR, PE, EPDM PVC, EVA, PT, SIR, FEP, PUR	Conductor Material: Cu, Ag, Ni, Ni-Cr-Alloy[10]
	Medium-Voltage Cables • Power-Supply	Insulation Materials: XLPE, PVC	Jacket Materials: PE, PVC	Conductor Material: Cu
Wires	all applications	Insulation Materials: PE, PVC, ETFE	Conductor Material: Cu	–

Figure 16 Overview of wire and cable materials at NPP Borssele

Step 2: Identification of Relevant Ageing Mechanisms and Long Term Environmental conditions

In this step, the technical design data of relevant materials is investigated to evaluate whether the environmental conditions to which the SCs are exposed may initiate significant ageing mechanisms. Moreover parameters of concern are assessed. This means that for each material a conservative long term temperature and radiation dose is determined. If relevant the sensitiveness of materials for other (environmental) conditions is described. For some materials and/or applications humidity and electrical field are detected as relevant.

This KCB specific review of ageing mechanisms is reported in the Catalogue of Ageing Mechanisms for Electrical Components. This report includes descriptions of potential ageing mechanisms, potential ageing effects and applicable stressors for component materials, as well as ageing relevant values specific for each material, serving as the basis for evaluating material behaviour.

An example of the report for the materials polyethylene, polypropylene and natural rubber is given in Figure 17.

Material	Ageing mechanism and effects
PE	Mechanisms: For polyethylene (PE), the most important ageing effects in air result from oxidative-degradation, caused by the attack of oxygen under thermal or radiological activation. The higher the temperature or radiation dose rate is, the faster these reactions occur (equal what medium the polymer reacts with). So the degradation rate of the material increases. With respect to ageing degradation, PE is more sensitive to normal thermal load than to normal levels of irradiation. Effects: Discoloration and embrittlement (and finally cracking) can occur when subjected to thermal or radiological exposure. Technical Data: A long-term temperature of 60 °C [16] and a radiation dose during long-term usage of 200 kGy [9] is evaluated for this material. Nevertheless, the equalized threshold value will be selected from those for PP (which is 50 kGy for the total dose).
PP	Mechanisms: The thermal stability of polypropylene (PP) is rather weak. The fastest ageing process is thermo-oxidative degradation caused by the attack of oxygen under thermal activation. The radiological stability of PP is moderate and more or less comparable to that of other polyalkenes like polyethylene. The higher the temperature or radiation dose rate is, the faster these ageing reactions occur. With respect to ageing degradation, PP is more sensitive to normal thermal load than to normal levels of irradiation. Effects: Discoloration and embrittlement (and finally cracking) can occur when subjected to thermal or radiological exposure. Technical Data: A long-term temperature of 100 °C [16] and a radiation dose during long-term usage of 50 kGy [9] are given for this material. Nevertheless, the equalized threshold values will be selected from those for PE (which is 60 °C for temperature).
NR	Mechanisms: For polyisoprene or natural rubber (NR), the most important ageing effects result from oxidative-degradation caused by the attack of oxygen under thermal or radiological activation. Because NR has partly unsaturated bonds, irradiation may lead to cross-linking. The higher the temperature or radiation dose rate is, the faster these reactions occur. With respect to ageing degradation, NR is more sensitive to normal thermal load than to normal levels of irradiation. Effects: Discoloration and embrittlement (and finally cracking) can occur when subjected to thermal or radiological exposure. Technical Data: A long-term temperature of 70 °C according to table 3.14 in [16] is given for this material. A limiting radiation dose could not be investigated. Nevertheless, the equalized threshold value will be selected from those for CR (which is 20 kGy for radiation).

Figure 17 Example of material evaluation at NPP Borssele

Step 3: Definition of Environmental Conditions

Based on design information, documentation of monitoring programmes and plant walk downs the relevant environmental conditions in the light of ageing were determined.

Environmental conditions during normal operation of KCB which can be relevant to ageing of cables and wires are:

- Temperature,

- Radiation, and
- Relative air humidity (moisture) / salty air in the outside area (oceanic climate).

Operational conditions are:

- Temperature (power cables),
- Vibrations, and
- Electrical fields (medium voltage).

During the plant walk down special consideration was committed to the presence of ultraviolet (UV)-rays, while a certain level of UV-radiation can cause ageing degradation to specific materials. However UV-rays have not been detected concerning passive electrical and instrumentation and control (I&C) components and therefore not regarded.

Mechanical vibrations can lead to failure of subjected equipment only within a relative short time and this should be detected during maintenance activities or periodic testing. The operating experiences gathered at KCB and other nuclear plants demonstrates that vibration has no influence on the function of passive long-lived electrical and I&C components important to safety.

The propagation of mechanical vibration from active components (e.g. rotating machinery) to the passive electrical and I&C components (like cables and wires) can be neglected in the light of ageing management. Such an effect does not affect adversely the ability of considered components to perform their intended function.

Electrical fields are an operational condition and affect only the cables that induce this stressor. Other components are not affected and therefore a definition of environmental areas regarding the electrical fields is not necessary. However, this stressor is considered in the framework of the Ageing Management Review (AMR).

Step 4: Identification of the Environmental Areas

To find out if cables, or more specific the applied materials, may be used in adverse environments, an overview of the environmental conditions in buildings or specific rooms was set up. In this case an adverse environment is defined as an environment where the temperature or dose rate exceeds the long term “design” parameters of the cable.

The environmental conditions in the environmental areas - which are described in the third column of the table (Figure 18) - are generally taken from the design basis conditions. When available, there is also a reference to measured values. The references used are defined in the fourth column of the table. If there are different values of environmental conditions described in the references the relevant ones are marked (underlined) in the third column. An example of the beneficial defined areas are shown in Figure 19.

Environmental area	Description	Environmental conditions	References used
outside area		**Design basis conditions:** temperature -12 °C to +30 ° C rel. air humidity 40 % up to 100 % salty air	Report Kraftwerk Union
building 01	reactor building, containment	**Design basis conditions:** temperature up to +50 °C [3] rel. air humidity ≤ 75 % **Measured values:** the environmental conditions within building 01 have been regarded within the environmental qualification program, the environmental data (temperature, radiation) was measured during a whole fuel cycle, see also table 3. Remark: The values found in the monitoring program are local values, in some cases hot spots in the specific room.	Report Kraftwerk Union Report AREVA
building 02	reactor building, annulus	**Design basis conditions:** temperature up to +35 °C rel. air humidity ≤ 60 % **Measured values:** the environmental conditions within building 02 have been regarded within the environmental qualification program, the environmental data (temperature, radiation) was measured during a whole fuel cycle, see also table 3. Remark: The values found in the monitoring program are local values, in some cases hot spots in the specific room.	Report Kraftwerk Union Report AREVA
building 03	auxiliary reactor	**Design basis conditions:**	Report Kraftwerk Union

Figure 18 Example of source documentation regarding environmental conditions, NPP Borssele

Environmental area	Environmental conditions	Rooms
building 01	temperature less then +40 °C radiation level not significant (≤ 1,57E-4 Gy / h)	
building 01	temperature up to +55 °C radiation level not significant (≤ 1,57E-4 Gy / h)	
	temperature up to +40 °C, radiation up to 0,1 Gy / h	
	temperature up to +55 °C, radiation up to 0,1 Gy / h	
building 01	temperature up to +72 °C, radiation up to 0,1 Gy / h	
	temperature up to +40 °C, radiation up to 0,6 Gy / h	
building 02	temperature up to +30 °C	rooms with vertical cable routing
building 03	temperature up to +55 °C	(hotspot conditions)

Figure 19 Example of defined environmental areas at Borssele NPP

Note that the environmental conditions are conservatively determined, most specific locations in the areas may have milder conditions.

Step 5 Assessment Regarding Environmental Conditions, Materials and Ageing Mechanisms

In this assessment materials were identified, which may be affected by significant degradation processes. It is based on a comparison between the local environmental conditions (see step 4 above) and the specific material parameter identified in the catalogue of ageing mechanisms (CAM)-electrical. As a result this assessment provides a table which includes a review of all the applied materials in cables regarding the defined conditions and shows whether the used material can be affected adversely by a potential stressor.

In Figure 20 and Figure 21 the results of the investigations in the CAM-electrical are summarized for all the passive electrical commodity groups. The identified stressors temperature, radiation, electrical field and moisture are linked to the used materials.

Material	Stressors			
	Temperature	Radiation	Electrical field	Moisture
CR	x	x	-	-
EP	x	x	-	-
EPDM	x	x	-	-
ETFE	x	x	-	-
EVA	x	x	-	-
FEP	x	x	-	-
FKM	x	x	-	-
MF	x	x	-	-
NR	x	x	-	-
SEBS	x	x	-	-
PA	x	x	-	x
PE	x	x	-	-
PE (medium Voltage application)	x	x	x	x
PEEK	x	x	-	-
PI	x	x	-	x
PO	x	x	-	-
PP	x	x	-	-
PPS	x	x	-	-
PT	x	x	-	-
PTFE	x	x	-	-
PUR	x	x	-	-
PVC	x	x	-	-
PVC (medium Voltage application)	x	x	x	x
SIR	x	x	-	-
TPC	x	x	-	-
XLPE	x	x	-	-

Figure 20 overview of materials and stressors (2) in NPP Borssele

Material	Stressors			
	Temperature	Radiation	Electrical field	Moisture
XLPE (medium Voltage application)	x	x	x	x
Ag	-	-	-	-
Al	-	-	-	-
Au	-	-	-	-
Cr	-	-	-	-
Cu	-	-	-	-
Fe	-	-	-	x
Ni	-	-	-	-
Sn	-	-	-	-
Zn	-	-	-	-
Cu-Alloy	-	-	-	-
Cu-Ni-Alloy	-	-	-	-
Cu-Sn-Alloy	-	-	-	-
Cu-Zn-Alloy	-	-	-	-
Ni-Cr-Alloy	-	-	-	-
Ni-Fe-Alloy	-	-	-	x
Sn-Ni-Alloy	-	-	-	-
Brazing/Soldering Alloys	-	-	-	-
Glass	-	-	-	-
Refractory Cement	-	-	-	-
Glass Wool	-	-	-	-
Mineral Wool	-	-	-	-

Legend:

x …important for AMR according to [8]

- …not important for AMR according to [8]

Figure 21 overview of materials and stressors (2) in NPP Borssele

Step 5a: Approach and Definition of Covering Comparison Parameter

General

To evaluate the potential ageing behaviour of the materials, the values of the environmental condition parameters in each relevant building have been compared with the material specific long term design values. Therefore covering comparison parameters (temperature and radiological dose) are defined based on measured and building design parameters. The result of this comparison is a determination whether the used material can be affected adversely by a potential stressor.

Definition of Covering Comparison Temperature (T_c)

If available, measured temperature values for a whole fuel cycle are taken into account because these reflect the local environmental conditions. Otherwise the building design temperatures inclusive a margin of 10 °C are applied.

It was found that in some cases the measured temperature values exceed the building design temperature values. The reason for this is that the temperature measurement has been performed direct at hot spots inside a considered room. These are locations/components which may be influenced by operating conditions like hot pipes.

The defined temperatures are listed up in the table below. T_c of 72 °C, 55 °C, 40 °C and 30 °C are measured values. T_c of 60 °C and 45 °C are the building design temperatures considering the margin of 10 °C.

An additional thermal stressor has to be considered for some cables due to self-heating. It is being caused by the electrical current. Therefore the self-heating temperature rise for components used in power applications is taken into account. For I&C application as well as for non-current-carrying components this effect can be neglected. For the comparison of these materials $T_{c'}$ was developed.

Building/Area	T_c [°C]	$T_{self\text{-}heating}$ [°C]	$T_{c'}$ [°C]	Temperature area
several rooms	72	13.9 / 16.4	85.9 / 88.4	Area T1
several rooms	55	13.9 / 16.4	68.9 / 71.4	Area T3
several rooms	40	13.9 / 16.4	53.9 / 56.4	Area T5
several rooms	60[19]	13.9 / 16.4	73.9 / 76.4	Area T2
several rooms	30	13.9 / 16.4	43.9 / 46.4	Area T6
all other rooms in building 02	45[19]	13.9 / 16.4	58.9 / 61.4	Area T4
several rooms	55	13.9 / 16.4	68.9 /71.4	Area T3
all other rooms in building 03	45[19]	13.9 / 16.4	58.9 / 61.4	Area T4
all other building/ rooms	45[19]	13.9 / 16.4	58.9 / 61.4	Area T4

T_c ... comparison temperature in accordance to table 2 and table 3
$T_{self\text{-}heating}$ ... increasing of temperature due to the self-heating (13.9 °C for all polymers except CR, NR, SIR, PP, and TPC; 16.4 °C for polymers CR, NR, SIR, PP, and TPC).
$T_{c'}$ ... comparison temperature with consideration of self heating

Figure 22 Overview of temperature areas, NPP Borssele

For the goal of the assessment 6 temperature Areas were defined, see Figure 22.

Definition of Covering Comparison Dose Rate (DR_c)

Exposure to radiation loads of passive electrical components is possible in building 01 (containment). This radiation was measured for a whole fuel cycle and therefore the measurement data reflect the local environmental condition. The measured value of the radiation dose is a hot spot dose in an area of the considered room. Figure 23 shows the determined dose rates DR_C regarding the 3 defined radiation areas.

Building/Area[20]	DR_c [Gy/h]	Source	Radiation area
several rooms	0.6	table 3	Area R1
several rooms	0.1	table 3	Area R2
several rooms	Radiation level not significant $\leq$ 1,57E-4	table 3	Area R3

DR_c ... determined dose rates in accordance to table 2 and table 3

Figure 23 Overview of radiation areas, NPP Borssele

Step 6: Comparison Regarding Thermal Conditions

General

The defined material design temperatures of all considered materials are compared with the defined comparison temperatures with respect to the different temperature areas.

Temperature area T1 (as an example)

Figure 24 shows the comparison of the defined comparison temperatures T_c and $T_{c'}$ for temperature area T1 and the material design temperatures T_d. The result of the comparison is shown in column “Assessment $T_d < T_c$ / $T_{c'}$”. If the result of the equation $T_d < T_c$ or $T_d < T_{c'}$ is ‘no’, than based on this approach significant ageing degradation in the LTO period is not expected. If on the other hand the result is ‘yes’, than the material may be critical for long term operation regarding ageing degradation and require dedicated AM.

This exercise is fulfilled for all 6 temperature areas.

Commodity group	Application	Material	T_c / $T_{c'}$ [°C]	T_d [°C]	Assessment $T_d < T_c / T_{c'}$
Cables	Low voltage	CR	72 / 88.4	70	yes / yes
		EPDM	72 / 85.9	90	no / no
		EVA	72 / 85.9	90	no / no
		FEP	72 / 85.9	150	no / no
		NR	72 / 88.4	70	yes / yes
		PE	72 / 85.9	60	yes / yes
		PI	72 / 85.9	260	no / no
		PP	72 / 88.4	60	yes / yes
		PT	72 / -	90	no / -
		PTFE	72 / 85.9	150	no / no
		PUR	72 / 85.9	90	no / no
		PVC	72 / 85.9	65	yes / yes
		SIR	72 / 88.4	200	no / no
		TPC	72 / 88.4	120	no / no
		XLPE	72 / 85.9	90	no / no
		Ag	72 / 85.9	not relevant	- / -
		Cu	72 / 85.9	not relevant	- / -
		Ni	72 / 85.9	not relevant	- / -
		Ni-Cr-Alloy	72 / 85.9	not relevant	- / -
	Medium Voltage	PE	- / 85.9	60	- / yes
		PVC	- / 85.9	65	- / yes
		XLPE	- / 85.9	90	- / no
		Cu	- / 85.9	not relevant	- / -

Figure 24 Example (Area T1) of comparison regarding thermal conditions, NPP Borssele

Step 7 Comparison regarding radiological conditions

General

In a similar approach as in the thermal assessment the design radiation dose of the materials are compared with the defined area comparison doses, in this case the cumulated dose for 60 operational years. Remark that this dose is just reality for the measured hot spot location of the area, for most components this dose is very conservative. Besides that a lot of cables are installed as part of modification programmes during the plant life and will have less than 60 years of operation.

Radiation area R2 (as an example)

The following table (in Figure 25) shows the comparison of the defined comparison dose D_{LTO} of 46 kGy and the maximum material specific radiation doses D_d for 60 years. The result of the comparison is shown in column “Assessment $D_d < D_{LTO}$”.

Commodity group	Application	Material	D_{LTO} [kGy]	D_d [kGy]	Assessment $D_d < D_{LTO}$
Cables	Low voltage	CR	46	20	yes
		EPDM	46	1000	no
		EVA	46	50	no
		FEP	46	5	yes
		NR	46	20	yes
		PE	46	50	no
		PI	46	2000	no
		PP	46	50	no
		PT	46	1250	no
		PTFE	46	5	yes
		PUR	46	20	yes
		PVC	46	200	no
		SIR	46	30	yes
		TPC	46	1250	no
		XLPE	46	1000	no
		Ag	46	not relevant	-
		Cu	46	not relevant	-
		Ni	46	not relevant	-
		Ni-Cr-Alloy	46	not relevant	-
	Medium Voltage	PE	46	50	no
		PVC	46	200	no
		XLPE	46	1000	no
		Cu	46	not relevant	-
Wires	All application	ETFE	46	300	no
		PE	46	50	no
		PVC	46	200	no
		Cu	46	not relevant	-

Figure 25 **Example (Area R2) of comparison regarding radiological conditions, NPP Borssele**

Step 8: Assessment Regarding Moisture

The assessment regarding moisture is not based on a comparison of material design values with environmental conditions because in general information about material design moisture is not available. A moisture environment is relevant only for material polyimide. For this material the humidity or salty air can lead to degradation processes. Polyimide is only used in the containment in thermocouple compensation cables for the in-core temperature measurement. Within the containment humidity or salty air is not a relevant stressor.

Step 9: Assessment Regarding Electrical Field

Electrical field as stressor for ageing degradation is in principle only relevant for medium voltage cables. In the case of low voltage cables the value of the electrical field is too low to cause degradation. The influence of only electrical fields regarding ageing can be neglected. The design parameters of SCs (e. g. operating voltage, thickness of insulation material) guarantees that the occurring field strength lies sufficiently under the critical electrical field strength. At the critical electrical strength an insulation breakdown occurs.

A second electrical degradation process in polymers occurs in the presence of electrical stress and moisture. These stressors lead to a development of water trees in polymers and consequently to a decrease of the dielectrical strength of an affected material causing finally an insulation failure. The assessment of water trees is not based on a comparison of material design values with environmental conditions.

Among other things an inhomogeneity of the insulation materials leads to this degradation process.

Because this inhomogeneity is not measurable with a reasonable effort, the development and impact of water trees is not quantifiable. This phenomenon is also only relevant for materials exposed to medium voltage, because in low voltage applications the value of the electrical field is too low.

Summarized Results of the Assessment

The table in Figure 26 shows the summarized results of the assessment of the considered stressors and their impacts on the materials. The x-marked cells except inside the columns “Electrical field” and “Moisture” show where the specific material design parameter is lower than the corresponding environmental parameter. The marks shown in columns “Electrical field” and “Moisture” mean, that these stressors may initiate significant ageing mechanisms.

Note: In the case of the materials PE, PVC and XLPE degradation process only occurs when the stressors “Electrical field” and “Moisture” are present simultaneously.

In temperature areas T1 till T6 it is to differentiate between components which are used in power application and in I&C application. In this case the first mark indicates the materials used in I&C applications and in non-current-carrying components. The second mark points out materials used in power applications.

Material	Summarized Results of Assessment									
	Temperature Area T1	Temperature Area T2	Temperature Area T3	Temperature Area T4	Temperature Area T5	Temperature Area T6	Radiation Area R1	Radiation Area R2	Electrical field	Moisture
CR	x / x	- / x	- / x	- / -	- / -	- / -	x	x	-	-
EP	- / -	- / -	- / -	- / -	- / -	- / -	x	-	-	-
EVA	- / -	- / -	- / -	- / -	- / -	- / -	x	-	-	-
FEP	- / -	- / -	- / -	- / -	- / -	- / -	x	x	-	-
MF	- / x	- / -	- / -	- / -	- / -	- / -	-	-	-	-
NR	x / x	- / x	- / x	- / -	- / -	- / -	x	x	-	-
PA	- / x	- / -	- / -	- / -	- / -	- / -	x	x	-	x
PE	x / x	- / x	- / x	- / -	- / -	- / -	x	-	-	-
PE (medium voltage application)	- / x	- / x	- / x	- / -	- / -	- / -	x	-	x	x
PI	- / -	- / -	- / -	- / -	- / -	- / -	-	-	-	x
PO	x / x	- / x	- / x	- / -	- / -	- / -	x	-	-	-
PP	x / x	- / x	- / x	- / x	- / -	- / -	x	-	-	-
PTFE	- / -	- / -	- / -	- / -	- / -	- / -	x	x	-	-
PUR	- / -	- / -	- / -	- / -	- / -	- / -	x	x	-	-
PVC	x / x	- / x	- / x	- / -	- / -	- / -	x	-	-	-
PVC (medium voltage application)	- / x	- / x	- / x	- / -	- / -	- / -	x	-	x	x
SEBS	- / -	- / -	- / -	- / -	- / -	- / -	x	-	-	-
SIR	- / -	- / -	- / -	- / -	- / -	- / -	x	x	-	-
XLPE (medium voltage application)	- / -	- / -	- / -	- / -	- / -	- / -	-	-	x	x
Fe	- / -	- / -	- / -	- / -	- / -	- / -	-	-	-	x
Ni-Fe-Alloy	- / -	- / -	- / -	- / -	- / -	- / -	-	-	-	x

Figure 26 Summarized results of the assessment, NPP Borssele

This assessment approach is not done on a quantifying base and does not deliver a remaining lifetime of SCs. The primary aim of this assessment is to focus all materials where a possibility of significant degradation processes due to the considered stressors and ageing mechanisms exist.

Step 10: Component based Assessment

General

Now the potential ageing behaviour of all the applied materials is investigated, in this step it is established which materials are applied in which thermal and radiological areas at the plant. For this purpose a database was develop in which all the relevant data was gathered. The database "LTO AMR component database" contains all relevant component specific information to carry out the assessment on the level of components. For the cables this means that every single cable was assessed. The following figure shows the general structure of this database:

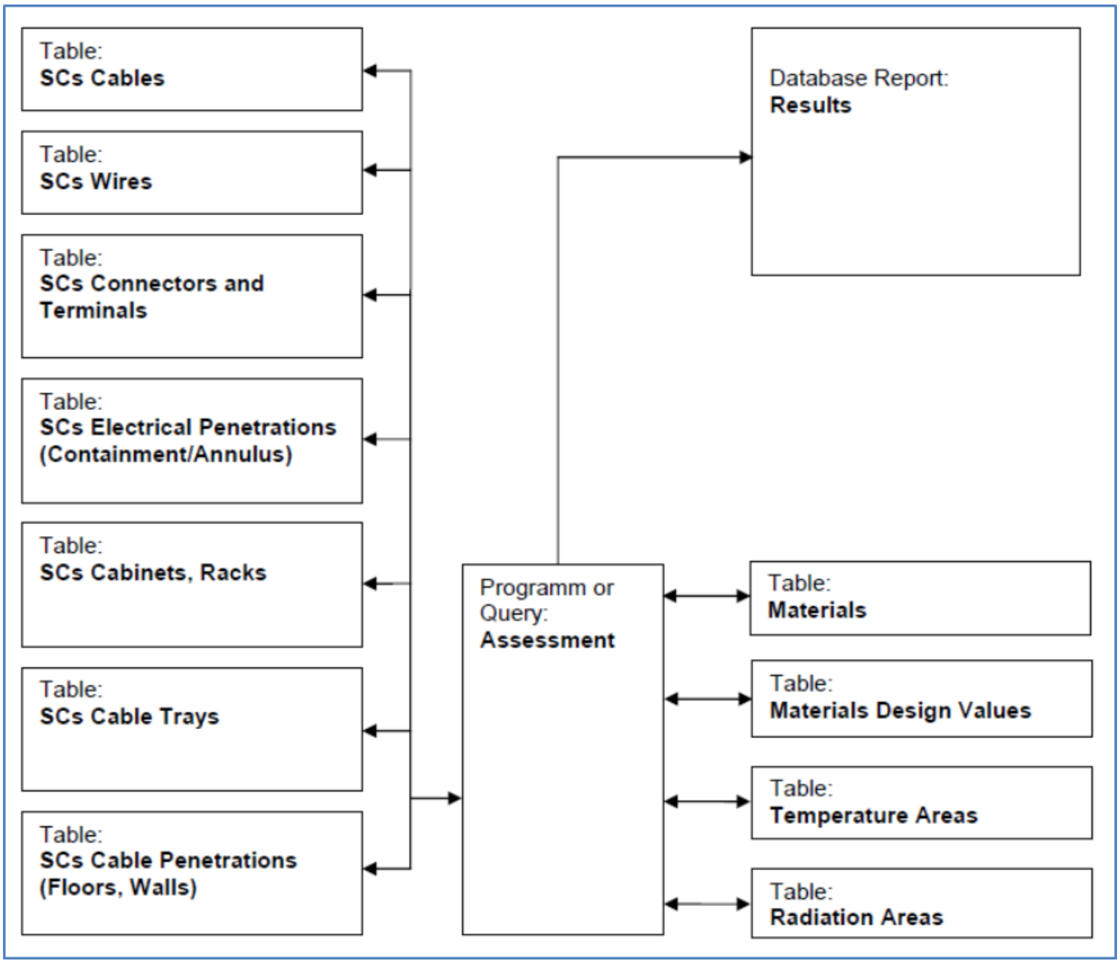

Figure 27 LTOB component database for NPP Borssele

Assessment for cables

In the result of the cable assessment it was found that 27 cables with a PVC insulation and jacket installed in the areas T1 or T2 may be critical regarding ageing degradation for the period of LTO. For another 15 cables installed in the areas T2 or T3 only the jacket may be critical.

Number of Cables	Material Insulation/Jacket	Temperature Area	Earliest Year of Installation	Note
27	PVC/PVC	T1, T2	1973	both materials are critical
15	XLPE/PVC	T2, T3	1973	only jacket material critical

Figure 28 result of the thermal assessment on single cable level, NPP Borssele

Regarding ageing degradation due to radiological load in radiation area R1 there are 27 cables with PCV insulation and jacket, and 43 cables with SIR insulation and jacket assessed as potential critical for LTO. Another 90 cables with SIR insulation and jacket are assessed in radiation area R2.

Number of Cables	Material Insulation/Jacket	Radiation Area	Earliest Year of Installation	Note
27	PVC/PVC	R1	1973	both materials are critical
43	SIR/SIR	R1	1973	both materials are critical
90	SIR/SIR	R2	1973	both materials are critical

Figure 29 Result of the radiological assessment on single cable level, NPP Borssele

Assessment for wires

Applied wires are insulated with PVC and PE and only wires used in the areas T1 and R1 are critical. Since in reality the wires are used in the spreader rooms and in I&C-cabinets only (which are not placed in the mentioned areas) a further evaluation is not necessary.

3.1.2.2 Ageing assessment of electrical cables - HFR-specific information

In the AMR the assessment of electric cables was part of the total electrical equipment and instrumentation assessment. The below mentioned ageing mechanisms were considered to be relevant for electrical SSCs:

- Material degradation due to radiation;
- Fatigue or wear which influences the conditions of electrical components attached;
- Corrosion effect caused by humidity;
- Corrosion of synthetic materials accelerated by external corrosion, temperature and radiation;
- Chemical processes in batteries, electrolytic capacitor and diesel oil of emergency generators.

For electrical systems, availability of components should be taken in account. Over long time frames spare parts may not be produced by manufacturers anymore. This kind of ageing defined as obsolescence does not influence the physical failure rate of a component. It may however extend the mean time to repair. All cables reviewed during the AMR have a good availability hence replacement will not cause problems.

Some cables will be subject to degradation by radiation. For all cables degradation of synthetic insulation material is considered to be relevant.

Damage was found on cables of nuclear instrumentation of which insulation material aged due to neutron radiation.

Previous analyses

In 2003 an investigation was performed into safety relevant cabling. This ageing assessment was re-evaluated in 2013 as part of the AMR.

Most cables appeared to remain in condition during operation. The only cables found subject to ageing were the nuclear instrument coax cables hence a preventive replacement programme has been established.

Surveillance programme electric materials HFR

In 2009 an evaluation aimed particularly on safety critical cables and connectors was executed. In addition an evaluation of the EIS-PPO's on ageing control was made.

3.1.2.3 Ageing assessment of electrical cables - HOR-specific information

The representative type of cable (as determined based on the selection criteria described above) of each category of electrical equipment is described in this section with respect to ageing. It should be realised that ageing stressors are of course present at the HOR, but to a lot smaller extend as when considering higher flux reactors or nuclear power plants. This is due to the open pool type design, where high temperature and pressures are not present. Also the operating power of 2 MW_{th} lowers the ageing stressor radiation, compared to reactors which are operated at higher power outputs. Ultraviolet (UV) light is not much of an ageing stressor at the HOR as all HOR related cabling run within the RID buildings (and not outdoors exposed to sun).

As was shown in the previous paragraphs category 1 till 3 are not applicable at the HOR. So this ageing assessment of electrical cables starts at category 4.

Category 4

The cables running from the on-site diesel generator to the low-voltage supply room and subsequently to the electrical cabinets in the containment are determined to be the most conservative representative in this category, due to their function and higher amperage application. The environment of these cables can be classified as mild also in an accident scenario as the cable routing is relatively far away and well screened from the reactor core.

Because of the relevance of these cables, the type of cable selected and year of installation was determined. It turned out that the cabling between the low voltage room and the electrical cabinets in the reactor containment, installed during the initial built of the reactor, was still in use. Based on this finding these cables have been replaced in the winter maintenance period of 2015-2016. Also the electrical power cabinets in the reactor containment were replaced in the same maintenance period, despite being of younger age while they were installed during the move of the control room in the 1980s.

The on-site diesel generator and cabling towards the low voltage supply room was replaced in 2000. The equipment in the low voltage supply room has been replaced in 2010. Due to the relative young age of these cables and the mild environment in both operating and accident scenarios, this cabling is, after the replacement of the old cables between low voltage supply room and the electrical cabinets in the reactor containment, fit for purpose.

Category 5

An ionisation chamber is used underneath the reactor bridge to measure dose rates and is selected to be the conservative representative of this category. Both the high voltage supply cable and the signal cable of ionisation chamber are super screened coaxial cables, manufactured to the Atomic Energy Standard Specification AESS (TRG) 71181 part 1 & part 2 “Super screened co-axial cables for the nuclear power industry”. The maximum allowed integrated radiation dose is specified to be 10^8 Rontgen for the applied MM10/75 type cable. These ionisation chamber and cabling were newly installed when the control room was moved. It can easily be shown that the accumulated radiation dose since then (~15 Sievert at the bridge dose rate monitor) is well below the specified value and these cables are therefore fit for use in both the operating and accident radiation environment. The maximum ambient temperature as specified for the applied type of coaxial cable is 85^{o}C. So well within the normal operating environment. In the specification, it is also stated that the MM10/75 cable can withstand at least 7 days of 100^{o}C, which would be about the temperature when the water in the pool would turn into steam in case of an severe problem with the cooling of the reactor core. So also with respect to operating/accident temperature the cable can be assessed to be fit for use.

To further enhance the post-accident monitoring capacities, it is currently investigated as an outcome of the Complementary Safety margins Assessment if a dedicated monitoring room, separate from the existing infrastructure, can be realised with the Reactor Institute Delft premises. Dedicated monitoring instruments within the reactor containment, separate cable routing and an independent back up power supply shall be used for this monitoring room.

Category 6

The selected safety channels are boron coated ionisation chambers in a waterproof housing. From the records it can be found that a lot of attention was given to applying the correct type of cabling to these detectors, especially within the waterproof housing as access to this cabling is not easy and radiation levels are elevated. Also the way to properly connect this cable is described in very detailed procedures. All this can be readily explained due to the fact that in the years previous to replacing these so called safety channels, there were a lot of problems with the cabling to the old detectors due to the accumulated radiation dose and connector problems.

The cabling currently in use is therefore the outcome of the experience gained when dealing with these problems. The cabling directly connected to the detector is so called mineral cable, known for its superior resistance to radiation. In the case of these detectors 1/8” stainless steel MgO insulated cables were used for both the high voltage supply cable and the signal cable, both are surrounded by ceramic fish spine beads for the first 2 meters. At 2m from the detector the ceramic fish spine beads are replaced by a peri-braid low boron content glass fibre sheathing which is soft soldered to the stainless steel MgO insulated cables at the transition from the fish spine beads to the glass fibre sheathing. Outside the waterproof housing the connectors are installed. This whole assembly was factory tested by dedicated test procedures. Outside the waterproof housing the mineral cable is connected to the super screened MM 10/75 cable also described in the previous category. The above cable assembly can be described as extremely rigid, especially for the environment to which it is exposed at the HOR. None of the ageing stressors present at the HOR at the location where this cable is applied (like radiation or temperature) are even close to the design specifications (119F0639 “Boron lined uncompensated ion chamber”) for this type of cable.

Category 7

Also the cabling connected to the fission chamber, as selected as the conservative representative of this category, is a mineral insulated cable, which is mated to an super screened coaxial cable (in this case the MM17/33 manufactured to the same AESS (TRG) 71181 part 1 & part 2 specification as the MM10/75 super screened coaxial cable) outside the waterproof housing surrounding the fission chamber. Just like described in the previous section this type of cabling is well suited for the environment to which it is exposed at the HOR.

In 2009 the fission chamber in operation did fail though due to the intrusion of moisture. The already existing instruction for flushing the fission chamber housing with instrument air has been reintroduced since that occurrence. This flushing was right after the installation of the new fission chamber part of the standard instructions, but was over the years considered to be less relevant as no problems were observed over long periods of not flushing the fission chamber housing.

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

3.1.3.1 Monitoring, testing, sampling and inspection activities for electrical cables - NPP Borssele

Note: Licensee EPZ has opted for combining the sections:

- Monitoring, testing, sampling and inspection activities for electrical cables - NPP Borssele (3.1.3.1)
- Preventive and remedial actions for electrical cables – NPP Borssele (3.1.4.1)

The licensee described the information as to be regarded as “Measures and activities regarding ageing management for the passive commodity group cables and wires”

This has been accepted by the ANVS.

Medium voltage cables, extended life time estimation using expert model

In 2002 an ageing assessment of all the 6000 V cables, safety relevant and non-safety, at the plant was executed by residual life calculations regarding thermal degradation with use of a model of KEMA (now DNV-GL). The model was adjusted to assess cables as critical at 60% of the theoretical thermal degradation of the cable. Only a few cables which are not important to nuclear safety were assessed as potentially critical to the end of the operational plant life at that time (2013).

Because of the LTO of the plant and the results of the LTOB-AMR assessments DNV-GL made an updated and more extensive remaining life analysis on all 6000 V cables at KCB in a so called desk study. Beside thermal degradation also water tree degradation was taken into investigation.

Based on the results the cables are grouped in 5 categories:

Table 5 Grouping of cables for remaining life analysis, NPP Borssele

Cable is in safe operation	Regarding the assessment no significant ageing degradation is expected before the end of LTO (2034)
Cable is operating in moderately safe range	The calculated end of safe life is between 2026 and 2034
Cable requires maintenance such as visual checks and diagnostic measurements soon	The calculated end of safe life is between 2017 and 2025
Cable is operating beyond its estimated safe end of life and should either be replaced and/or undergo diagnostic tests	The calculated end of safe life is between 2008 and 2016
Cable is operating a considerable period beyond its estimated safe end of life. Without further knowledge, this cable should be regarded as critical. Replacement and/or testing is highly recommended	The calculated end of safe life is < 2008

In the calculation model the safe end of life is based on certain maximum allowable degradation. For the insulation material XLPE 80% is chosen, for PVC this is 90%. It must be noted that the model for XLPE is based on a lot of more knowledge and experience as for PVC, as PVC is much less applied for medium voltage cables. For these reasons the model is more accurate for XPLE cables.

A second important issue is that the thermal calculation is based on ageing degradation behaviour of materials and cables, whereas the data for water treeing is based on worst case failure data of cables. Especially for PVC insulated cables the model is based on limited experience. For example the safe life regarding water treeing for PVC cables is set to 19 operational years. Nevertheless 25 cables at KCB are operational since the commissioning of the plant in 1973, and no water tree faults are occurred until today. Five of these cables are buried in soil and connect components in the cooling water inlet building with the power busses.

The calculation results of the 56 cables are summarized in the table below:

Table 6 Calculation result for 56 cables, NPP Borssele

	Thermal			Water treeing		
	Safety	Safety related	Non safety	Safety	Safety related	Non safety
safe operation	5 (XLPE)	3 (XLPE)	1 (XLPE)	13 (9 x PVC) (4 x XLPE)	3 (PVC)	3 (PVC)
moderately safe range	-	12 (PVC)	9 (PVC)	1 (XLPE)	-	-
requires maintenance	6 (PVC)	1 (PVC)	2 (PVC)	1 (XLPE)	-	-
beyond its estimated safe end of life	1 (PVC)	2 (PVC)	7 (PVC)	5 (PVC)	-	3 (2 x PVC) (1 x XLPE)
considerable period beyond its estimated safe end of life	-	-	7 (PVC)	5 (PVC)	3 (1 x PVC) (2 x XLPE)	19 (PVC)

The concerns regarding thermal ageing degradation are focused on the cables with PVC insulation. There are no cables important to safety characterized as critical, but some economically important cables are.

Regarding water treeing the results show a larger number of cables that are beyond the estimated safe end of life according to the model. But as stated before the knowledge in the model regarding water treeing is limited.

<u>Medium voltage cable: replacement of cables, field and laboratory measurements</u>

With the finish of the desk study an approach to perform the next steps in ageing management on medium voltage cables was developed:

1. In the outage of 2017 the cables of the house load transformer are replaced.
2. Field tests will be performed before the cables of the house load transformer are removed from the cable trays, especially Partial Discharge and Dielectric Spectroscopy. On the one side these measurements give information about the momentary quality of the cable insulation, at the other side the results of the measurements serve as correlation for the results of the laboratory tests (step 3).

3. Laboratory tests on the replaced cables of the house load transformer. The laboratory test will give a detailed view of the quality of the insulation. The state of thermal degradation will be clear and also the presence of water trees in the insulation.
4. Based on the gathered information with the measurements in step 2 and 3 the remaining life model will be updated. And even more important it is expected that a measurement programme for the remaining cables can be developed, which includes a correlation between measurement results and the estimated remaining life of the cables.
5. If necessary the replacement programme will be extended, based on the results of step 2, 3 and 4.

Low voltage cable and wires: ageing management activities

Detection of ageing degradation before it leads to malfunctioning of electrical circuits is the main goal of ageing management of cables and wires. The activities are mainly based on the results of the LTOB project, extended with measures due to operating experience.

The strategy to manage ageing of cables and wires can be summarized in the following steps:

1. The focus is on the safety cables and wires which are due to their materials, operating experience, environmental conditions and/or process conditions potentially sensitive for ageing degradation.
2. Visual inspections are carried out on a broad scope of cables and wires to detect signs of ageing degradation in an early stage. Non-safety cables and wires are taken into account as well, because this may give useful operational experience.
3. Testing of mechanical and/or electrical properties when gathering qualitative data of the condition of cables and wires is necessary.
4. Replacement of cables and wires if the results of the inspections makes it necessary.
5. Monitoring of the environmental conditions in the containment with the goal to react in a timely manner on an eventually change of temperature or radiation level which may have influence on the ageing degradation rate of cables and wires.
   Note: also other conditions, e.g. humidity in cable conduits, belong to the monitoring programme.
6. A specific on-going qualification programme for cables with a design base accident requirement (see chapter 3.1.2.1.1 ‘Qualified life assessment for cables with a functional requirement during design base accidents’).

<u>Visual Inspections</u>

Based on the results of the LTOB-AMR visual inspections are carried out to detect discoloration, cracking due to embrittlement, melting and swelling of wire and conductor insulation and cable jacket. Also leakage of plasticizer out of insulation is a point of attention.

Wiring in the spreader rooms

Visual inspection

Wires on the nuclear and conventional spreaders are visually inspected on a yearly base.

Simplified elongation at break test

Yearly a selection of spreader wires is tested in a simplified elongation test, in which the elongation-at-break of 0,5 is checked by winding the wire around its own diameter. If cracking is detected, than the necessary of extended measures will be determined by engineering judgement.

Laboratory elongation-at-break tests

Laboratory elongation-at-break tests, including accelerated ageing, of spreader wiring is performed every 5-years. The scope of the programme is based on the results of earlier tests and the experiences of the measures described above. If results of tests make it necessary the time period will be decreased.

Wires in electrical cabinets

Visual inspection

Yearly the wiring in the electrical cabinets is inspected. Signal wiring as well as wiring of the electrical power is involved in the inspection. If the results of the inspection give rise, then additional measures will be formulated.

Wires and cables of actuators

Visual inspection

Inspection of wires and cables, and connectors, of actuators is carried out in combination with other maintenance activities on actuators.

Low voltage cables (general)

Visual inspection

Visual inspection of cables is integrated in revision and inspection activities on (active) components like low voltage bus bars, actuators, cable bridge (incore instrumentation and control rod actuation), electrical motors, instrumentation, etcetera.

Replacement of cables

Low voltage cables of actuators (example).

Actuators are connected with a power and a signal cable with a length of a few meters and connectors. It was found that the original SIHF cables have a mechanically weak silicon rubber jacket. This may lead to damages, probably followed by affection of the insulation and therefore malfunctioning of the electrical circuit. These cables are replaced for mechanically stronger FRNC-2G2G and FRNC-JE-L cables.

More general are cables replaced in modernization projects, by which a major part of the safety relevant cables are a shorter time in operation as the plant itself.

Neutron flux instrumentation cables

Measurement of electrical insulation resistance

Yearly insulation resistance measurement of the neutron flux detector cables are carried out. This measure belongs to the third group of cables as defined in the WENRA technical specification.

Deep well pump cables

Measurement of electrical insulation resistance

Yearly insulation resistance measurement of in soil cables of the deep well pumps are carried out. This measure belongs to the second group of cables as defined in the WENRA technical specification.

Cable conduits

Inspection environmental conditions

Every 6 years an inspection of cable conduits will take place, in which is controlled if the conduits are dry. This measure belongs to the second group of cables as defined in the WENRA technical specification.

Monitoring environmental conditions

Within the LTOB project an extensive programme to monitor the environmental temperature and radiological loads in the several buildings was carried out. As an increase of environmental conditions may accelerate ageing degradation, the monitoring programme will be repeated in 2026 to check the stability of the environmental conditions.

Low voltage cable

Laboratory investigation long term behaviour: Analysis and prediction of the reliable operational life of non-LOCA cable in the scope of the result of the AMR assessment.

A conclusion of the AMR assessment is that the quality of cables with PVC insulation or jacket may be critical for long term operation in the environmental areas T1, T2 and T3, as described in chapter 3.1.2.1. Operational aged cables in the steam relieve valve room are replaced to serve as test specimen to determine a reliable operational lifetime. The cables in the steam relieve valve room were selected because of the relative high environmental temperature of approximately 45 °C and the operational life of the cables.

The reliable thermal operational lifetime of 3 test cables will be determined by DNV-GL in line with standard IEC 60502-1. To simulate another 40 operational years the cables will be artificial aged at 100 °C for 46 weeks. Every two months the mechanical properties of jacket and insulation material are measured and compared with a failure criterion.

The next step in AM for this group of cables depends on the results of the laboratory test.

Cables with XLPE or SIR as insulation or jacket may be critical for long term operation in the radiological areas R1 and R2. Although cables with a radiological load as defined as the conservative comparison load were not localized, a set of cables in the containment are replaced to serve as test specimen to determine a reliable operational lifetime at the locations with the highest measured radiation levels.

The analysis programme to predict a lifetime will be about the same as for the thermal loaded cables, with the exception that these cables also will be artificial aged with a radiological load.

3.1.3.2 Monitoring, testing, sampling and inspection activities for electrical cables - HFR-specific information

In order to monitor the condition of cables visual inspection and test are being executed.

The visual inspections are included in the EIS PPO system. Results are reported and if needed corrective maintenance action are taken.

To gain a better understanding of the cable ageing mechanisms in practice a cable depot has been introduced in 2014. The cables listed in section 3.1.1.2 ’Scope of ageing management for electrical cables - HFR-specific information’, are stored in this depot. Test samples are taken and subjected to radiation in the HFR pipe corridor and temperature conditions exceeding the normal reactor operation conditions. Results are recorded and used for improvement of the PPO and inspection plans. If necessary cables will be preventively replaced.

First results of this programme did not show obvious degradation. The programme will possibly be extended and improved during development of the extended maintenance concept (see chapter 2 and its HFR-specific sections).

3.1.3.3 Monitoring, testing, sampling and inspection activities for electrical cables - HOR-specific information

Electrical cables can basically be split up in instrumentation cables and power supply cables. As was shown by the categorization of the electrical equipment, electricity is not a requirement for safely shutting down the reactor, nor to cool the reactor after shutdown or to isolate the reactor environment from the ambient environment in case of an accident. Therefore, power supply cables are not continuously monitored, as their failure will show in the category 5 till 7 instruments that they are powering.

Instrumentation channels are by (correct) design continuously self-monitoring. Digital outputs are so by their fail safe design and the output of an analogue channel will be a 4-20mA (or equivalent) signal. In case the channel fails to supply a current within this range the instrumentation channel itself or the monitoring station will generate an alarm.

Furthermore at the HOR a check out is performed on the instrumentation and electrical systems every Monday morning by default. By executing such a weekly checkout a functional test of the cabling is performed.

Next to the activities described above lower frequency inspections to low voltage equipment is formalised in following the NEN-3140 Operation of electrical installations – Low voltage. By following this norm it is ensured that all legal requirements for operating installations till 1000VAC and 1500VDC are met. For some of the subsystems of the HOR this inspection is performed by a certified inspector of the supplier of this subsystem. The NEN-3140 inspections are carried out not only for the equipment related to the HOR, but also for all scientific instrumentation as present at the Reactor Institute Delft.

All the cabling within the HOR is readily accessible which facilitates inspection and corrective action if required.

3.1.4 Preventive and remedial actions for electrical cables

3.1.4.1 Preventive and remedial actions for electrical cables - NPP Borssele

Note: Licensee EPZ has opted for combining the sections:

- Monitoring, testing, sampling and inspection activities for electrical cables - NPP Borssele (3.1.3.2);
- Preventive and remedial actions for electrical cables – NPP Borssele (3.1.4.2).

The information can be found in section 3.1.3.1.

3.1.4.2 Preventive and remedial actions for electrical cables - HFR-specific information

Within the EIS PPO programme PPOs for cable maintenance are available. The preventive maintenance actions are scheduled by maintenance work preparation. Results are recorded and if necessary corrective maintenance action are taken. The results are also being used for further preventive maintenance programme development.

The below listed PPOs are available for preventive maintenance on cables.

Table 7 PPOs available for preventive maintenance on cables at the HFR

System/ component	Type of preventive maintenance	Frequency	PPO number
Cable samples pipe tunnel (corridor below reactor pool)	Cable monitoring, visual inspection cable samples subject to radiation and temperature	3 yearly	EIS-PPO-S06/01 A
Cable samples pipe tunnel	Monitor cable ageing	2-yearly	EIS-PPO-S07/01 A

Visual inspections as executed according the related PPO, are executed by good electric technician craftsmanship. When any deviation from normal conditions is noticed, a stretch/ bend test will be executed. This test will be performed according to the CEI IEC 811-3-1 part 3; 'Methods specific to PVC; Test for resistance to cracking'.

3.1.4.3 Preventive and remedial actions for electrical cables - HOR-specific information

As can be seen from the categorization of the electrical equipment, electricity is not a requirement for safely shutting down the reactor, nor to cool the reactor after shutdown or to isolate the reactor environment from the ambient environment in case of an accident. Therefore a safety oriented maintenance strategy could rely on a replace-on-failure strategy. But to ensure the reactor can be used as effective as possible it proofed well worth the determine the age and ageing stressors of the cabling towards the electrical equipment based on the electrical equipment categorisation and environment. Especially replacing the old power supply cables between the low voltage supply room and the electrical cabinets in the reactor containment was a direct consequence of this analysis. Another example of a preventive action is the flushing of the fission chamber housing by a constant supply of instrument air to avoid the accumulation of moisture.

In order to further minimize the impact of ageing cabling, all cables tags were checked at the start of the cable ageing management. Whichever cable tag was not clear anymore was replaced by a new standardised type of cable tagging. The tagging itself of course does not help to improve the performance of the cable, but proper cable tagging will speed up the identification of the individual cables and therefore troubleshooting.

To further enhance the post-accident monitoring capacities, it is currently investigated as an outcome of the Complementary Safety margins Assessment if a dedicated monitoring room, separate from the existing infrastructure, can be realised with the Reactor Institute Delft premises. Dedicated monitoring instruments within the reactor containment, separate cable routing and an independent back up power supply shall be used for this monitoring room.

3.2 Licensee's experience of the application of AMPs for electrical cables

3.2.1 Licensee's experience of the application of AMPs for electrical cables - NPP Borssele

AM of passive electrical commodity groups is primarily handled by a dedicated working group, wherein employers of various departments are represented. In this cooperation the tasks of the departments were coordinated and attuned to each other. The working group preliminary focus on the passive electrical commodity groups. However ageing experience on active components is discussed as well, although the management of active components is handled in the preventive maintenance programme. For a single unit plant, and the only NPP in the country, this is a practical and workable approach with a high amount of efficiency. Because of LTO proper ageing management is become even more important as it was already. With the introduction of a formal procedure the development and organization of AM at KCB is increased.

Some experiences related to Ageing Management of electrical cables:

<u>Leakage of plasticizer out of wire insulation</u>

In an inspection of an electrical cabinet a greasy substance on electrical PVC isolated wiring was found. A subsequent study learned that migration of plasticizer out of the PVC insulation caused this. In a broader inspection of electrical cabinets there were found some other spots. Since than an increased periodical inspection plan of the electrical cabinets, and the located places in particular, is carried out. The quality of the affected wires is a main aspect in this inspection plan.

<u>Protection of medium voltage cables against ultraviolet rays</u>

During a course of medium voltage cable the sensitivity of red coloured 6000 V cable sheets for UV-rays (due to sunlight) was a topic. UV-rays may degrade the material leading to cracks and because of that to degradation of the mechanical protection of the insulated conductors. As a measure the cable sheets of the start-up transformer were protected with UV-ray protection tape.

<u>Embrittlement of cable insulation</u>

During the outage of 2003 embrittlement of insulation of the cable of a control rod drive was detected. The cable is replaced and the ageing management programme is improved to ensure early detection of degraded cable condition.

Another example of a replacement of cables in the past due to embrittlement of insulation are the cables of the pressure vessel heating in 1983, so after 10 years of operation.

Valve drive cables close to hot pipes

During a walk around after an outage of the plant some valve drive cables were found mounted to close to hot pipes. In spite of clear instructions the cables weren't mounted properly after the revision of the valves. The inspection as part of the revision programme is improved to prevent cables for improper mounting.

Ageing degradation of cables of temperature transmitters

In a visual inspection signs of ageing degradation of the insulation of a cable of temperature measurement of a main coolant pump bearing was found. After a comprehensive inspection on similar cables in similar environmental conditions it was decided to replace the cables.

Wiring in pressure transmitters

An external operating experience report of potential ageing degradation of wiring insulation in pressure transducers leads to a specific inspection point in the periodic maintenance activities of this type of transducers.

Summarized: the experience with the described approach of AM is quite positive. Ageing degradation seems to be managed well, there are no major problems due to unexpected ageing degradation of passive electrical components.

3.2.2 Licensee's experience of the application of AMPs for electrical cables - HFR-specific information

There is an ageing management programme for electric cables. Ageing of the cables as kept in the cable depot is managed adequately.

The AMR reports recommendations for improvement of the existing programme in order to remain in control in the future. This means the preventive maintenance and inspection programme has to be extended to make sure all safety critical cables are monitored sufficiently at all times.

The recommendation will be implemented by means of the RCM programme NRG started in 2017.

At the HFR most cables hardly showed signs of degradation. The most ageing sensitive cables where found in the reactor pool. This concerns coax cables to nuclear instrumentation. As the cables are subject to neutron radiation the cables showed obvious degradation. As a result a preventive measure for replacement is established.

The most safety relevant cables at the HFR are considered to be the underground emergency power supply cables running from the independently working emergency generator sets to the HFR. These cables are redundant and its functioning is monitored by special units, for which spares are available. Yet a full inspection programme in order to determine degradation does not yet exist. Development of such an inspection will be part of the ageing management programme improvement.

3.2.3 Licensee's experience of the application of AMPs for electrical cables - HOR-specific information

Every deviation at the HOR is captured in a deviations database. This deviations database was consulted for cable related deviations. Cabling ageing cannot be traced to be the reason for many deviations as reported in the deviations database over the last decade. There are several good reasons how this can be explained.

Nearly all cables at the HOR have been replaced in the years 1980-1982 when the control room was moved from within the reactor containment to outside of the reactor containment. During this replacement of instrumentation and associated cabling it can be clearly found in the records that experience gained over the earlier years of operating the HOR reactor was used to make sure correct types of cables were selected, especially for those exposed to raised radiation levels. Some of the power supply cables were not replaced at that time. But those have also been replaced in more recent years, based on the findings during the development of the cable ageing management programme as part of the general ageing management programme.

Furthermore ageing stressors, like high temperature, pressure or radiation levels, are a lot less severe at an open pool type reactor operated at 2MWth like the HOR, when compared to different reactor designs or nuclear power reactors.

There was a noticeable exception to the general observation that cabling ageing is not a major source for deviations. This was the moisture intrusion into the fission chamber cabling in 2009. This experience has been used to reintroduce the flushing of both the fission chamber housing in operation and the spare one.

As shown in this chapter cabling is not safety critical at the HOR. Therefore a corrective maintenance strategy can be applied to cabling at the HOR. While setting up the cable ageing management programme the age of the various cables in use for the HOR reactor were evaluated. The electric cables running from the low voltage room to the reactor building, which were installed during the original build of the reactor, were replaced. Cables are functionally checked at the HOR and replaced when they no longer function or cause a malfunction.

The cable ageing management programme has also raised awareness on selecting the correct cable for its application. This is a valuable lesson for the future OYSTER project at the HOR, as many new cables will have to be selected. A special section on cabling is therefore added to the design requirements document for the HOR.

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

3.3.1 Regulator's assessment and conclusions on ageing management of electrical cables - NPP Borssele

Electrical cables AMP is structured as all other AMPs of the NPP. It was assessed during the LTO-project, including GRS support and IAEA SALTO missions. Based on that a licence modification was granted.

Cables relevant for LOCA or HELB conditions are handled in a special cooperation programme under VGB. Outcome of the LTO-programme was the replacement of some electrical equipment, including cables. A lot of international cooperation is going on with the other KWU plants. A cable deposit is used in one of the German plants. Another solution will be necessary after the German phase-out.

The other (non-LOCA) cables and wires underwent a structured 10-step approach to determine the cables and wires critical to ageing degradation, taking into account all possible influences. It turned out that only about 200 cables were critical. For wires no critical ones were found. Visual and other inspections and test programmes are focused on these cables. Visual inspections also take non-safety related cables for additional experience. Test programmes are aimed at determining and following the actual amount and development of degradation. The environmental conditions, that were the basis of the assessment of critical cables are periodically monitored to determine significant change.

This AMP was under scrutiny during the LTO-project. It is considered to be effective.

Further refer to section 2.7.

3.3.2 Regulator’s assessment and conclusions on ageing management of electrical cables - Commonalities at all RRs

Both research reactors have carried out an AMR and described the AMP for cables and wires. At the HFR in 2014 was introduced a cable depot. This might be a candidate for a good practice for a research reactor. In general not much degradation was found, but several cables have to be included in the programme. The most ageing sensitive are the coax cables to the neutron flux measurements in the reactor pool. They will be replaced as a preventive measure. The licensee has found that the functioning of the redundant underground power cables from the diesel generators to the HFR is monitored, but the cables themselves are not yet under a degradation monitoring programme. The safety importance of these cables makes this necessary. At the HOR there is a different situation, because the HOR does not need electricity for the preservation of the three main safety functions. A corrective maintenance strategy is therefore applied. Many cables were replaced 35 years ago when the control room was moved to the outside of the reactor building, newly specified cables taking account of the environmental circumstances were installed that still function very well. After the AMR several other cables have also been replaced. In 2009 moisture intrusion into the cable for the fission chamber was detected and remedied by a periodic action to flush the housing.

It can be concluded that in general the ageing of cables of the research reactors is under control.

Further refer to section 2.7.

4 Concealed pipework

This chapter only applies to NPP Bossele and the HFR in Petten.

4.1 Description of ageing management programmes for concealed pipework

4.1.1 Scope of ageing management for concealed pipework

4.1.1.1 Scope of ageing management for concealed pipework - NPP Borssele

Due to the design of the Borssele NPP (KCB), the plant has very few nuclear safety relevant concealed pipework. Part of the auxiliary and emergency cooling water system (system VF) is composed of concealed composite concrete/steel piping, so-called Bonna-pipes, buried in soil. This system provides service water for cooling of SSCs that are important to safety. The Bonna-pipes connect the sea site structures of the intake and outlet buildings with the plant's cooling water systems and are therefore subjected to sea water.

The back-up residual heat removal water cooling system (System VE) uses buried glass fiber reinforced epoxy (GRE) pipes. Eight pumps in this system can in severe accident conditions provide groundwater from wells on site amongst others to heat exchangers in the back-up residual heat removal system (TE), heat exchangers in the nuclear spent fuel storage pool cooling system (TG) and to cool the water basins of the back-up emergency feed water system (RS). The discharge of the cooling water is via the discharge line of the Bonna-piping of the main cooling water system or through the sewerage system.

The low pressure fire extinguishing system (UJ-system) also largely consists of concealed pipes. Except for the fire extinguishing capacity, this system may also be used for:

- providing an alternative water supply to the TE-system in case of unavailability of the VE-system under severe accident conditions;
- extending the availability of the emergency and back-up emergency feedwater systems in severe accident conditions;
- maintaining pressure on the VE-system while its own pumps are not in use.

The buried pipes of the UJ-system consist of GRE piping, Polyethylene (PE) piping, and partly steel piping.

The buried pipes of these three systems are concealed and thus relevant for this Topical Peer Review (TPR).

The pipes from the radioactive waste water treatment system are not concealed at KCB. These pipes connect to the VF-system, so that any radioactive effluents containing concealed pipes are covered by the ageing management of the VF-system.

No pipes for the transfer of diesel fuel for emergency power generation are concealed.

The concealed Bonna-piping of the VF-system consists of a steel core, with a thick cover of concrete on both the in- and outside. Except in the case of physical damage to the outside of the concrete cover, the steel core is always protected from the environmental conditions related to the soil that the piping is buried in. The VF-piping is exposed to soil on the outside, sea water on the inside, and where it is not buried, to outdoor air, with temperatures ranging from -10°C, up to 50°C.

Relevant ageing mechanisms for these components have been established in accordance with the Borssele NPP ageing management procedure[35] which was described earlier for Borssele NPP in 2.3.1.1, 2.3.2.1, and 2.3.3.1.

The concealed piping part of the VF-system was replaced with new Bonna piping during the refuelling outage of 2012. The reason for replacing the piping of this system was to change the lay-out of the cooling systems to improve the design basis by improved application of the Redundancy, Diversity and Independence ('RDI') concept on these systems. This could be regarded as addressing conceptual ageing, due to the increased requirements with regard to RDI over the years. The routing of the VF-system was physically separated from the pipes of the main cooling water system. Although the piping is now quite new to be subjected to ageing management for Long Term Operation, all ageing mechanisms are equally valid for the old and the new situation, and so is the required management thereof.

4.1.1.2 Scope of ageing management for concealed pipework - HFR-specific information

The following parts of the installation include concealed pipework:

- Primary cooling water piping;
- Pool cooling water;
- Hot and warm water drain;
- Contaminated water distribution to decontamination and waste Treatment (DWT);
- Compressed air distribution.

These are addressed below.

Primary cooling water piping in concrete

The reactor core is cooled by the primary cooling water system. Part of the primary cooling water lines is concealed in concrete of the reactor pool and another part is surrounded by jacket pipes. Leak water in the jacket pipe system is monitored with a dedicated leak detection system and generates an alarm in the control room. In addition, the room humidity is also monitored from the control room.

Pool cooling water system

The pools are cooled by means of a separate cooling water system. A limited part of the cooling water lines is concealed by concrete.

Hot and warm drain lines

A part of the reactor pool hot and warm water drain lines are concealed by concrete.

Pipe line to decontamination and waste treatment (DWT)

Drained contaminated water is distributed to the DWT by a pipeline made out of HDPE. The pipe line is a soil buried line which is surrounded by a jacket pipe. Drain wells containing leak detection will collect water in case of pipe line leakage.

[35] PU-N12-50, Ageing management process

Compressed air distribution lines

Compressed air is used for air instruments, air drive control systems, air requiring personal protective equipment PPE, potable water system pressure, demineralize system charge air, instrument drying. A limited number of the distribution lines is concealed by concrete.

4.1.2 Ageing assessment of concealed pipework

4.1.2.1 Ageing assessment of concealed pipework - NPP Borssele

Ageing mechanisms that are relevant to the concealed pipework and that require ageing management activities have been established in accordance with the method as described in chapter 2 of this national assessment report (NAR) on the overall ageing management programme requirements and implementation. This method is established in the living ageing management programme of the plant[36] and makes use of catalogues of ageing mechanisms for mechanical[37] and structural[38] components. All ageing mechanisms that have been considered for this purpose were identified after careful consideration of the operating conditions, material properties and environmental conditions of the VE-, VF- and UJ-systems as described in the process description in chapter 2, in accordance with the guidance provided in the catalogues of ageing mechanisms for mechanical and structural components[37,38]. The guidance in these catalogues of ageing mechanisms was compiled, based on the latest insights and state-of-the-art in ageing mechanisms, by experts from the original equipment manufacturer (AREVA).

Research was performed in the last 40 years on several materials used to manufacture Nuclear Safety System components and subcomponents. Relevant research results for the ageing mechanisms of materials in the scope of this assessment are referenced in the catalogue of ageing mechanisms for mechanical components (CAM-MC)[39], with respect to Long-Term Operation of KCB.

The catalogues also provide comprehensive lists of reference documents, from IAEA, US-NRC, KTA, VGB R&D, proceedings from international conferences, research overviews, independent books, etc., ranked for individual ageing mechanisms. The CAM-MC lists a total of 186 references in this way.

The process by which KCB maintains its internal knowledge by participation in R&D projects is described in the overall chapter on the ageing management process in chapter 2.

The relevant ageing mechanisms for concealed piping are summarized in Table 8. This table lists the relevant SSC or part thereof, the material of its construction, and describes its environmental conditions. Based on this information, relevant ageing effects were identified, and the ageing mechanisms that would cause these ageing effects. The last column of the table lists the programmes that are conducted to manage the relevant ageing mechanisms.

Concealed VE-piping is all made of Glass Fiber Reinforced Epoxy (GRE). According to the state of scientific and technical knowledge of AREVA NP[37], GRE exposed to temperatures up to 50°C and Raw Water or Treated Water is not susceptible to corrosion, wear and plate-out. GRE is also resistant to ultraviolet radiation and nuclear radiation. Therefore, no relevant ageing mechanisms are identified

[36] PU-N12-50, Ageing management process
[37] PU-N12-50-101, Catalogue of ageing mechanisms, mechanical components
[38] PU-N12-50-103, Catalogue of ageing mechanisms, structural components
[39] PU-N12-50-101, Catalogue of ageing mechanisms, mechanical components

for GRE exposed to the environmental conditions it is applied to in its current application. The inside of the pipes is very smooth, which prevents the build-up of deposits. Sand and mud may precipitate due to low flow rates, but this effect will be negated during higher velocity flow rate testing.

Table 8 Ageing Measures for concealed piping (VE, VF and UJ), NPP Borssele

SSC	Material	Environment	Ageing Effect	Ageing Mechanism	Ageing Measure at KCB
Cooling water piping	Concrete covered carbon steel	Air – Outdoor Raw Water, Flowing Soil/Groundwater Buried	Loss of Material	• Freeze – Thaw • Erosion, Abrasion or Cavitation • Aggressive Chemicals • Corrosion of Embedded Steel and Steel Reinforcement	VF- lines → 3-Yearly visual inspection of the auxiliary and emergency cooling water system
			Cracking	• Freeze – Thaw • Reaction with Aggregates	
			Change in Material Properties	• Leaching of Calcium Hydroxide • Carbonation	
Cooling water piping	Concrete covered carbon steel	Soil/Groundwater Buried	Loss of Form, Cracking	Settlement of soil in which the piping is buried	VF-lines → 5-Yearly setting survey of the auxiliary and emergency cooling water system buildings and pipe lines
Cooling water piping	Carbon Steel	Embedded in Concrete Raw Water Flowing Soil/Groundwater Buried	Loss of Material	• Steel is attacked only when concrete is chipped away (damaged)	VF-lines → 3-Yearly visual inspection of the auxiliary and emergency cooling water system
			Cracking	• Stress Corrosion • Intergranular Corrosion	
Cooling water piping	Rubber/ Polymer	Outdoor air, Raw Water Flowing Soil/groundwate	Change in Material Properties, Loss of Weatherproofing/Integri	• Freeze – Thaw • Cracking, porosity and	VF-lines → 3-Yearly visual inspection of the auxiliary and emergency cooling

SSC	Material	Environment	Ageing Effect	Ageing Mechanism	Ageing Measure at KCB
		r	ty – Cracking	visible stress damage, • Corrosion	water system
Extinguishing water piping	Carbon Steel	Soil/Groundwater Buried	Loss of material	• General corrosion due to dissolved oxygen	UJ-lines - Monthly functional testing, and - monitoring of any leakage by trending the periods of running for the jockey pump
Cooling water piping, Extinguishing water piping	Glass Fiber reinforced Epoxy	Soil/Groundwater Buried	None identified	None identified	VE-lines, UJ-lines - Monthly functional testing, and - monitoring of any leakage by trending
Extinguishing water piping	Polyethylene (PE)	Soil/Groundwater Buried	None identified	None identified	UJ-lines - Monthly functional testing, and - monitoring of any leakage by trending the periods of running for the UJ jockey pump

4.1.2.2 Ageing assessment of concealed pipework - HFR-specific information

Relevant aging mechanisms

The following ageing mechanisms have been evaluated:

- Degradation by radiation: This is not relevant for concealed piping of the HFR;
- Temperature ageing: This is not is not relevant for concealed piping of the HFR;
- Creep, stress and relaxation: This is not relevant for concealed piping due to the relatively low temperatures in the system;
- Wear/ damage/ fatigue: Damage or fatigue could be caused by (abnormal) vibration of pumps. The configuration and connection of the concealed pipes will make any heavy vibration of lines impossible. Therefore this is not considered to be an ageing mechanism to be managed;
- Corrosion: Apart from most of the primary cooling water lines and the distribution line to the DWT concealed pipes are not covered by a jacket pipe. These pipes are protected by a taped layer but corrosion is still considered as a viable ageing mechanism.

In 2012 NRG found tritium leaking from a buried cooling water transfer pipe. The pipe had only a taped layer and was severely damaged by corrosion. This pipe had no leak detection, nor was it on the maintenance list. This incident led to the need of an improved AMP which includes pipe line condition monitoring. The transfer pipe line itself has been replaced by a new none-buried pipe, surrounded by another pipe, with leak detection. Hence it is not a concealed pipe line anymore. Currently all other relevant underground pipes are being part of a monitoring programme.

Primary cooling water piping in concrete

The jacket pipe drying system is connected to the leak detection system. Besides leak detection on concrete concealed pipes the drying systems prevents condensation in the void space between the pipe line and jacket pipe. This will avoid corrosion of pipes.

Pool cooling water system

A part of the pool cooling water lines is concealed by concrete. These pipes are considered to be subject to corrosion caused by interaction with concrete.

Hot and warm drain lines

The hot and warm drain lines covered by concrete can be subject to corrosion.

Pipe line to decontamination and waste treatment (DWT)

The pipe line to the DWT is covered by a jacket pipe. The pipe line will not corrode by water from soil as it is protected by the jacket pipe. In addition timely corrective maintenance is possible without loss of containment to the local environment. The leak detection system will indicate the occurrence of any leaking water.

Compressed air distribution lines

The compressed air lines concealed by concrete are subject to corrosion. The system supplies air to the air instrument systems. Loss of air due to leaking lines is considered to be limited. In addition the air instruments have instrument air compressors delivering redundancy. As a result the risk of pipelines leaking is limited.

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

4.1.3.1 Monitoring, testing, sampling and inspection activities for the concealed pipework - NPP Borssele

The Borssele NPP In-service-Inspection programme is based on the Dutch Nuclear Safety Guide NVR-NS-G-2.6, the 2007 ASME XI Code for safety class 1, 2 and 3, and national regulations concerning steam and pressure equipment. The buried and underground tanks and pipes (BTP) are not part of the ASME-programme. The inspections of BTP are covered by civil engineering codes, such as shown in the following box:

NEN 3398, Buitenriolering – Onderzoek en toestandbeoordeling van objecten
NEN 3399, Buitenriolering – Classificatiesysteem bij visuele inspectie van objecten
NEN EN 13508-1, Toestand van de buitenriolering – Deel 1: Algemene eisen
NEN EN 13508-2 Toestand van de buitenriolering – Coderingssysteem bij visuele inspectie

To assess the civil SSCs of the Borssele NPP, inspections that are described for relevant items in the component-oriented maintenance programme at the plant are evaluated against the ageing management requirements that were imposed by the identification of relevant ageing mechanisms.

Resulting from these inspections, the condition of the relevant SSCs is reported in inspection records. The responsible engineer evaluates the results and initiates further investigations or measures, if considered necessary. When dealing with Bonna-piping, the plant's civil department always consults an external specialist and together, they determine the correct measures for any issues detected during these consultations.

Because the BTP in the scope of this assessment do not fall under the requirements of the European Pressure Equipment Directive (PED), no Certified Third Party (CTP) such as Lloyd's Register is involved.

Table 9 SSC-based inspection programmes for VF004/005 Bonna-piping, NPP Borssele

Programme	**Interval**
Visual inspection of auxiliary and emergency cooling water piping VF004/VF005	3 years
Settlement survey of cooling water lines VF	5 years
Performing concrete repairs on auxiliary and emergency cooling water piping	As required

The programmes consist of periodic inspection to manage the effects of Loss of Material, Cracking and Change in Material Properties and activities to prevent impact on their intended function by ageing effects such as Loss of Form and cracking due to settlement and flow (shifting) of the soil in which the pipes are buried.

Visual inspection to ensure that the inner concrete cover is intact is an effective method to ensure that corrosion of the embedded carbon steel pipe material has not occurred and that the intended function is maintained. The measures according to the programme for visual inspection of the auxiliary and emergency cooling water system consist of regular camera inspections of the inner concrete surface and flexible connections.

Visual inspection of auxiliary and emergency cooling water piping

The in-/exterior concrete cover for cooling water piping experiences the ageing effects Loss of Material, Cracking and Change in Material Properties. The visual inspection programme of the auxiliary and emergency cooling water system VF004 and VF005 is applicable to manage these ageing mechanisms.

The inspection includes a 3-yearly video inspection performed by a qualified contractor. More detailed visual inspections are recommended if the inspection results indicate that further investigations are necessary. While inspecting the condition of the concrete layer, the internal steel layer is examined only where damage to the concrete is identified. The internal piping sheet will then be exposed for further investigation by removing the damaged concrete. Ageing effects found or any damage to the SSCs important to safety are mapped and recorded in the inspection report according to their size and nature.

No visual inspection from the outside is performed for the concealed cooling water piping. There are no known applicable ageing mechanisms for the concrete outer surface under these environmental conditions. Any leakage would take place through the connecting sealing members, which are covered by the internal visual inspections that take place every three years. The good external condition of the VF-pipes that were replaced in 2012 confirmed this supposition.

<u>Sealing member</u>

The inspected rubber parts seal the connection between the cooling water pipe line and the rigid steel flange for the intake building. They are exposed to outdoor air, flowing sea water and soil/groundwater. Special attention is given to “Change in Material Properties” and “Loss of Weatherproofing/Integrity – Cracking” with respect to cracking of the sleeve material, porosity and visible stress damage, corrosion of the collar ring and corrosion of the bolted connections. The tightness of the bolts and the fastening torque are also examined. Another aspect is the rubber material, which must comply with the manufacturer’s requirements for shore hardness. This is measured in accordance with the relevant DIN-standard[40].

Criteria for immediate replacement of the sealing member are determined by the particular damage that has occurred, such as any visible cracks, leakages or tears, or the measurement of a shore hardness value (according to the DIN standard) of the rubber where the manufacturer recommends replacement. Trending of the measured hardness does not take place, because the recommended shore hardness provides sufficient margin for the efficiency of the seal to remain operational between inspection intervals.

<u>Settlement survey of cooling water lines VF</u>

For determination of any settlements of the cooling water lines VF, the level of the lines is measured and compared with the as-built condition on a 5-yearly basis. Detected deviations are recorded in a geodetic levelling programme, using vertical tubes or access facilities to give admission to the benchmarks of this VF-piping.

<u>Leakage detection</u>

Leakage in the UJ-system is detected by monitoring the number of starts of the jockey pump that pressurizes the system. With any leakage in the system, the frequency of jockey pump starts would rise. This parameter is trended in the plant’s control room.

While the VE-system is not in operation, it is kept under pressure by the UJ-system, to prevent air ingress, which could otherwise enter through vacuum conditions in the elevated parts of the system. Due to this open connection to the UJ-system, monitoring the starts of the UJ-jockey pump would also indicate any leakages in the VE-system.

Groundwater measurements are taken on a yearly basis at different areas around the NPP grounds. The purpose of these measurements is to determine any leakage containing Tritium, which would probably involve the VF-system, because pipes containing radioactive effluents enter into the VF-

[40] DIN 53 505, ‘Härteprüfung nach Shore A’

system. Furthermore a radiation measurement device is installed in the VE-system to detect leakage in the heat exchangers of the fuel pool cooling system or the back-up residual heat removal system.

4.1.3.2 Monitoring, testing, sampling and inspection activities for the concealed pipework - HFR-specific information

Primary cooling water lines

The primary cooling water lines are inspected according the in-service inspection programme. This programme is established to monitor the condition of the pipe lines. It is based on the ASME guidelines for power reactors and reviewed and approved by Lloyd's Register.

The programme describes the method and frequency for visual and volumetric weld inspections. The reducers of the inlet pipe lines are inspected at the connecting welds at the beginning and end of the pipelines not being covered by concrete.

Elements described below are:

- Leak detection system
- Jacket pipe drive system
- Inspection before start

Leak detection system

A jacket pipe drying system controls the air condition of the primary cooling water pipe lines surrounded by the jacket pipes. This will avoid corrosion of the pipe lines.

Possible leak water is distributed by the jacket pipes. This leak water is collected and automatically detected. Detected leak water will automatically generate pipe isolation and activate an alarm in the control room. By separately isolating each pipe the leaking pipe can be detected. The AMR concludes the pipes will always leak before break as a result of the operation conditions. As a result possible pipe failure will always be detected in time and corrective maintenance actions will be taken.

The leak detection system is installed in the reactor hall basement. Whenever water is detected in the jacket pipe air drying system this may be caused by leaking concrete concealed pipe lines. Activity measurement of collected water will determine whether its primary cooling water form a leakage or water from another source. Eventually a notification will be triggered to the AMS.

The base condition for this system is similar to outside air hence corrosion is considered to be an ageing mechanism.

The leak detection system is functionally tested before start-up. Daily visual inspection will make sure defect components are timely notified. This combination of activities is sufficient to manage ageing effects.

Jacket pipe drying system

The air dryer (JD-01-ADR) is being checked daily. Temperature and humidity are recorded (BVH07). When the jacket pipe drying system detects a leakage, an alarm is activated which will result in pipe isolation.

Leak detection of primary leak path are tested on a regular basis. This operational inspection is a functional start test. Leakage trends are registered in the data acquisition system.

Inspection before start

Before each reactor start the complete HFR installation is inspected visually concerning leakages and other deviations. Components are inspected at the following locations:

- Primary Pump Building Cells;
- Primary Pump Basement;
- Reactor Hall Sub Pile Room;
- Reactor Hall Basement;
- Reactor Hall Pipe Corridor.

Remarks found for each location are recorded. Although this procedure will not detect ageing mechanisms, in an early state it will detect abnormal conditions caused by ageing.

Pool cooling water system

Non concealed pipes are inspected regularly. The PPO M009A for instance includes inspections on the pool cooling system before a reactor start. Concealed piping is however often very hard to impossible to inspect. A programme has started to group the concealed piping of the pool cooling system based on operational conditions and ageing mechanism. Selected pipes from each group that are accessible for inspection, will be examined and will provide condition assessments for the remainder of the pipes.

Hot and warm drain lines

The PPO PPO-M-009A includes regular inspections on drain lines before each reactor start. Currently there is no dedicated ageing management programme for this system. When the previously mentioned inspection programme on pool cooling water lines is finished, it will be extended with the hot and warm drain system.

Pipe to decontamination and waste treatment (DWT)

The condition of the pipe to the DWT is monitored by leaking water noticed in the drain wells. The jacket pipe delivers mitigation for any leaks which may occur. The inspection for leaks is described in PPO PPO-M-009. This is a weekly inspection.

A leak pressure test is executed each 5 years as described in PPO PPO-M-100.

Compressed air distribution lines

After HFR reactor shut down before each start a system inspection according PPO PPO-M-009A is executed. During this inspection components unusual conditions and deviations are being recorded. Any leaking pipelines will be heard. Necessary corrective actions will be taken. During the AMR process conditions were found not to deliver sever ageing mechanisms hence degradation of pipelines will be noticed in time.

4.1.4 Preventive and remedial actions for concealed pipework

4.1.4.1 Preventive and remedial actions for concealed pipework - NPP Borssele

No actions have been defined for preventing ageing of the concealed VF cooling water pipes. All ageing prevention that is necessary, is accomplished by its design. The design was proven reliable when the old piping was replaced due to conceptual ageing, as explained in the description of the scope for concealed pipework.

There are also no preventive actions for the concealed parts of the low pressure fire extinguishing system and backup residual heat removal water cooling system, which consist of GRE and PE piping, for which no relevant ageing mechanisms are applicable.

When it is identified that corrective action is required, a dedicated corrective action programme (remedial) for the concerned UJ, VE and VF pipelines is applicable.

4.1.4.2 Preventive and remedial actions for concealed pipework - HFR-specific information

Primary cooling water lines

ISI results

The results of the in service inspection programme will be used for corrective maintenance actions if needed.

Jacket pipe drying system

The air dryer (JD-01-ADR) and ventilation (JD-01-PMP) are subject to a yearly preventive maintenance programme executed by a third party (technical service company).

The dryer and ventilator are being maintained on a yearly base. Daily records of relevant parameters are made. In addition daily visual inspections are made. Before each reactor start-up another operational inspection is executed.

The jacket pipe drying system is a preventive system which has to function properly. Modifications have recently been made in order to avoid jacket pipe and pipeline degradation by corrosion. Malfunction of the system is noticed on time by daily inspections. The dryer (JD-01-ADR) and ventilator (JD-01-PMP) need limited maintenance; maintenance is managed by a service agreement.

The preventive maintenance, inspection and detection of malfunction or defects lead to timely replacement of components. As a result there is sufficient ageing management on the system.

Pool cooling water system

When inspections show degradation of pipes additional wall thickness measurement will be taken.

Hot and warm drain lines

A preventive maintenance programme is established for pumps, valves and instruments of the hot drain system. Degradation of these components will indicate the condition of pipelines as well. If necessary corrective maintenance action will be taken on both components and pipes.

Pipe to decontamination and waste treatment (DWT)

As the pipe is covered by a jacket pipe and subject to a test and inspection programme additional preventive actions on ageing are not needed.

Compressed air distribution lines

The system is maintained according related PPOs. The PPOs describe the preventive maintenance actions on compressed air components and pipes. Any corrective action needed on not concealed pipes will indicate the need of replacement of concealed pipes.

4.2 Licensee's experience of the application of AMPs for concealed pipework

4.2.1 Licensee's experience of the application of AMPs for concealed pipework - NPP Borssele

The assessment in this chapter is based primarily on engineering judgment through the comparison of ageing mechanisms and existing ageing management activities at the plant.

Using a combination of national and international expert's knowledge, KCB developed a civil ageing management programme between 2003 and 2008, by now having almost ten years of experience in running the programme. The different inspections, set with the help of external experts, consider the experience of operating plants and the safety requirements for relevant components.

The slide-in connections of the VF-piping sections were designed to handle influences of tidal cycles in terms of settlement. An ageing related event that has occurred at KCB since the plant was commissioned consisted of leakages of the VF-piping. These leakages resulted from a modification during the early years of operation, when it was decided to apply a rigid design for the connection of the piping to the cooling water inlet building rather than a flexible design. The connection between the steel piping leading to the inlet building and the concrete piping started leaking due to the lack of flexibility in this connection and resulted in slow degradation of the concrete connection due to tensile stresses. The concrete piping and steel connection, in both loops were returned to the original design of flexible connections, which solved the problem. The connections in the newly replaced Bonna piping were also equipped with the flexible connection design.

An existing ageing management measure is considered adequate, if it is technically suitable to detect the ageing mechanism of concern with an acceptable probability and if an inspection interval has been chosen to detect the ageing effect early enough (considering experience of degradation rates). Therefore, no additional AMPs were required for the concealed piping without applicable ageing mechanisms, such as for the concealed piping of the UJ and VE systems.

As the actual activities that are required for effective ageing management of the concealed piping have been employed since plant commissioning, their purpose is well understood and accepted by all relevant staff. Technical meetings and regular training update sessions are adequate measures to sensitize personnel.

The existing inspection programmes are determined as sufficient and adequately managed so that the intended function(s) will be maintained consistent with the KCB licensing basis for a service period of 60 years, i.e., Long-Term Operation.

4.2.2 Licensee's experience of the application of AMPs for concealed pipework - HFR-specific information

There is an ageing management programme for concealed pipelines. Ageing of the concealed pipe lines surrounded by a jacket pipe is managed sufficiently. Pipe drying systems and leak detection control systems aid to monitor the pipe line condition. All concealed pipelines directly related to safe reactor operation (reactor criticality / shut down control, reactor cooling and containment) are covered by a jacket pipe. Inspection programmes and preventive maintenance actions are executed for the important safety critical concealed pipes.

In addition an extended preventive maintenance and inspection programme is being established for concealed pipe lines without jacket pipes. These pipelines are not directly related to reactor safety The pipelines however need to be managed by an extended maintenance programme in order to avoid any future internal containment leakages.

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

4.3.1 Regulator's assessment and conclusions on ageing management of concealed pipework - NPP Borssele

The amount of safety relevant concealed piping at the NPP is not very high. These systems are the auxiliary and emergency cooling water system (system VF), the back-up residual heat removal water cooling system (System VE) and the low pressure fire extinction system (system UJ), that also has some functions in (severe) accident management. No piping of diesel fuel or radioactive water are concealed. After replacement of relevant parts of VF due to conceptual ageing in 2012 the ageing prevention is delivered by the design, confirmed by inspection programme. Ageing prevention of VE is also controlled by design and by regular flushing during periodic testing. The monitoring programme completes the ageing management. The AMP is considered to be effective.

Further refer to 2.7.

4.3.2 Regulator's assessment and conclusions on ageing management of concealed pipework - HFR-specific information

With regard to the research reactors, concealed piping only is applicable to the HFR. There is a number of concealed pipes that has leak before break design and associated detection mechanism. A number of pipes are partly inaccessible and therefore sometimes difficult to inspect or do not have jackets. The ageing management programmes for these are now under development. In a buried tritium transport pipe a leakage was discovered in 2012. This pipe had no leak detection, nor it was on the maintenance list. It has been replaced by a pipe above ground. Also this case was one that was a wake-up call for increased activity to improve ageing management. No conclusion can be drawn now because the extended AMP is not yet finished.

Further refer to 2.7.

5 Reactor pressure vessels

Although the HFR does not feature a pressure vessel in the usual sense, its licence holder NRG has volunteered to address this issue to some extent. The contribution of NRG can be found in Annex A. This chapter addresses the NPP Borssele only.

5.1 Description of ageing management programmes for RPVs

5.1.1 Scope of ageing management for RPV & Scope of ageing management for RPVs - NPP Borssele

General introduction Reactor Pressure Vessel NPP Borssele

The RPV of NPP Borssele has been designed by Siemens (KWU) and delivered by the former Dutch company Rotterdamsche Droogdok Maatschappij NV. A schematic view of the RPV including the internals can be found in Figure 30.

Design parameters:

- Design pressure: 175 bar(g)
- Design temperature: 350°C

The normal operating pressure is 155 bar, the RPV inlet temperature is 292,5°C and the RPV outlet temperature is 317.5°C.

As a basic material low-alloy steel 22NiMoCr3-7 was used complying with ASTM A508 Class 2. On the inside of the RPV a two-layered austenitic cladding is in place.

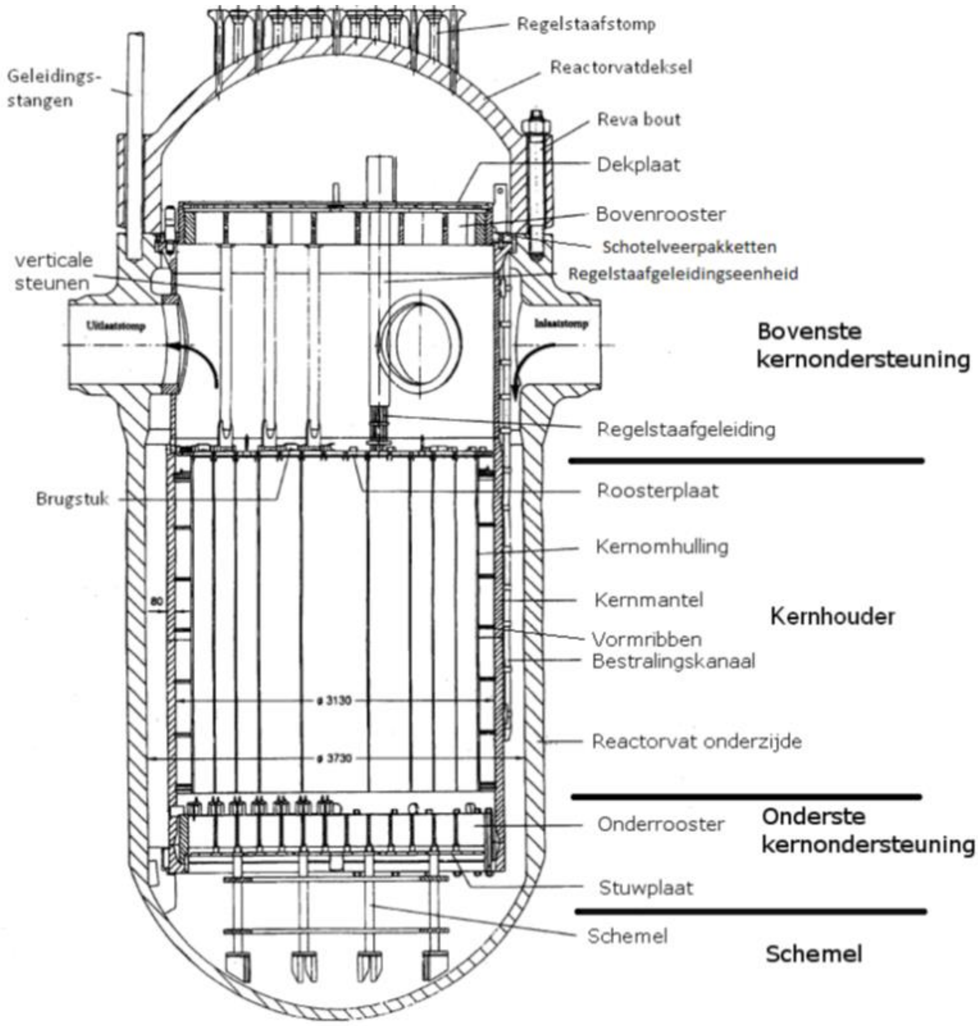

Figure 30 Cross section of KWU RPV

5.1.2 Scope of ageing management for RPV

NPP Borssele is a two loop plant and therefore the RPV has two inlet and two outlet nozzles.

In comparison with other designs the KWU RPV design has among others the following specific features:

- no nozzles or guide tubes within the lower part of the RPV;
- special screwed design for the control rod drive and instrumentation nozzle penetrations (because of this design no Nickel-based dissimilar weld is in place);
- a relative big water gap between the fuel elements and the vessel wall resulting in a relative low neutron fluence on the vessel wall;
- materials used with relative low sensitivity for neutron induced embrittlement.

Ageing Management Review of the RPV

To prove safe Long Term Operation (LTO) of NPP Borssele an assessment was performed based on IAEA Safety Report No 57[41]. This is described in NRG-report NRG-22701/10.103460[42]. Part of this assessment was scoping & screening to determine all systems, structures and components (SSC) important or relevant to safety. A description of the scoping & screening process can be found in chapter 2 of this NAR.

The RPV came out of the scoping & screening process as one of the passive highly safety important components which are not allowed to fail. Therefore a comprehensive AMR was performed on it.

The scope and the results of the AMR were used to implement an umbrella type ageing management process (described in chapter 02). The ageing management of the RPV is documented in the Integral Management System[43] of the plant. The important steps of the strategy are dealt with in 5.1.2.

Scope of the RPV ageing management programme

In Figure 31 a cross section of the RPV is given with the subcomponents belonging the scope of the RPV AMP. Note: the studs of the RPV head also belong to the scope but are not visible in the figure.

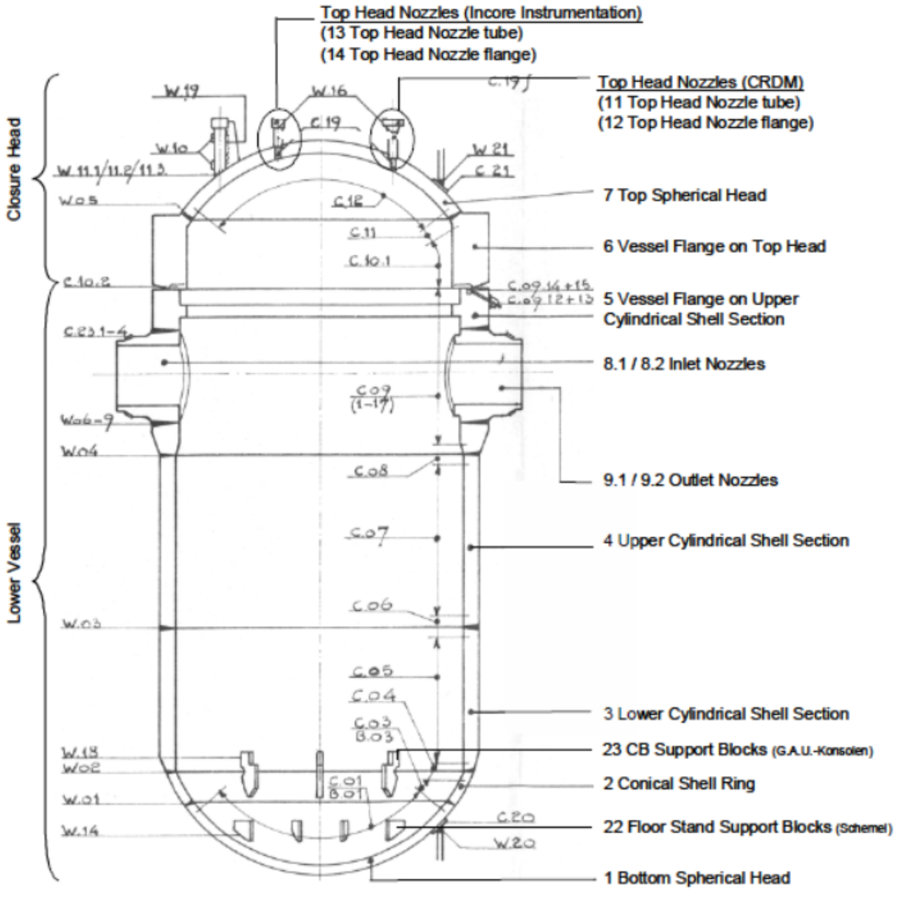

Figure 31 Cross section of RPV of NPP Borssele, with the subcomponents belonging the scope of the RPV AMP

[41] Safe Long Term Operation of Nuclear Power Plants, IAEA Safety Reports Series No. 57, 2008
[42] Conceptual document LTO “Bewijsvoering” KCB, NRG-report NRG-22701/10.103460, 9 September 2011
[43] Verouderingsstrategie reactorvat, EPZ Uitvoeringsprocedure PU-N12-50-303, 9-3-2016

In Table 10 all the subcomponents together with the materials are given.

Table 10 In-scope subcomponents and materials (the numbers in the first column refer to figure above), NPP Borssele

Reactor Pressure Vessel			
Item no	quantity	description	material
1-9	11	Main forgings (cylindrical parts and plates)	Low-Alloy Steel (22NiMoCr 3 7)
11	33	Top Head Nozzle tubes (CRDM)	External: low-alloy steel (St. 52)
13	10	Top Head Nozzle tubes (Incore Instrumentation)	Internal: Austenitic Stainless Steel (X10CrNiNb 18 9)
12	33	Top Head Nozzle flanges (CRDM)	Austenitic Stainless Steel (X10CrNiNb 18 9)
14	10	Top Head Nozzle flanges (Incore Instrumentation)	Austenitic Stainless Steel (X10CrNiNb 18 9)
n/a	6	Incore Instrumentation Assembly	Austenitic Stainless Steel (X10CrNiNb 18 9)
n/a	2	Level Measurement Assembly	Austenitic Stainless Steel (X10CrNiNb 18 9)
66	4	Blind Flanges of reserve Incore Instrumentation Nozzles	Austenitic Stainless Steel (X10CrNiNb 18 9)
21	3	Blind Flanges of reserve Level Measurement Nozzles and center CRDM Nozzle (including the fasteners)	Austenitic Stainless Steel (X10CrNiNb 18 9)
29	1	Venting Nozzle	Austenitic Stainless Steel (X10CrNiNb 18 9)
15, 16, 17	45	Closure Head Studs, Nuts, Washers	Low-Alloy Steel (AISI 4340 comparable to 40 NiCrMo 8 4)
22	16	Floor Stand Support Blocks (Schemel)	Austenitic Stainless Steel (X10CrNiNb 18 9)
23	6	Core Barrel Support Blocks	Austenitic Stainless Steel (X10CrNiNb 18 9)
73	4	Core Barrel Guide Blocks	Nickel-Base Alloy (Inconel 600)
19	4	RPV Support Blocks (external RPV support structure)	Low-Alloy Steel (22NiMoCr 3 7)

5.1.3 Ageing assessment of RPV

R&D

Research programmes played an important role in developing RPVs with high safety margins against failure during the complete operating lifetime and taking into account ageing stressors like high energy neutron flux. In the USA in the sixties the Heavy-Section Steel Technology (HSST) programme started. In Germany several research projects started. Important features of the RPV of NPP Borssele are the result of the research programmes. For a comprehensive overview of the R&D of RPVs and particularly German R&D reference is made to a German publication[44].

To adequately assess the ageing of SCs, NPP Borssele specific Catalogues of Ageing Mechanisms were developed by the OEM (Original Equipment Manufacturer) as part of the AMR. This catalogue comprises information regarding all ageing mechanisms identified to be of potential relevance for KCB Mechanical SSCs. This catalogue is used as a living document in the current ageing management process. See also description in chapter 2 of this NAR.

[44] Reaktorsicherheit für Leistungskernkraftwerke, Paul Laufs, Springer Verlag 2013

In the AMR of the RPV the ageing mechanisms were identified which could potentially result in significant degradation of the in-scope (sub)components. In Table 11 these ageing mechanisms are given and the subcomponents of the RPV for which they are applicable.

In the AMR the following information is used to assess the applicable ageing mechanisms:

- Manufacturing, design and specifications
    - For example welding processes and heat treatment
- Environmental and operating conditions
- Operation and maintenance history
    - For example inspection results
    - Generic nuclear industry experience and relevant research results, for example relevant ageing issues at other plants.

Table 11 Relevant Ageing Mechanisms for the RPV of NPP Borssele

Relevant Ageing Mechanisms	Affected Components and Subcomponents
Neutron-Induced Ageing (Irradiation Embrittlement)	RPV Core Region (Cylindrical Shell Section)
Ageing under Cyclic or Transient Loading (Fatigue)	All RPV Subcomponents
Borid Acid Corrosion	External Surfaces of Low-Alloy Steel including Mechanical Fasteners
Pitting Corrosion	Internal and External Surfaces of Low-Alloy and Austenitic Stainless Steel
Crevice Corrosion	Internal and External Surfaces of Low-Alloy and Austenitic Stainless Steel
Primary Water Stress Corrosion Cracking (PWSCC) of Nickel-base Alloys	Core Barrel (CB) Guide Blocks including the welds
Corrosion Fatigue (Environmental Fatigue)	All RPV Subcomponents

In the following for all the above mentioned ageing mechanisms the assessment and management are described.

Neutron-induced Ageing (Irradiation Embrittlement)

A well-known ageing mechanism applicable to most RPVs of PWRs is embrittlement due to fast neutron irradiation. Due to embrittlement the vessel can lose its ductility. In case of the existence of flaws and a significant loss of ductility severe loading could finally result in brittle failure of the vessel.

According to the applicable safety standards high safety margins must be shown to be sure that the risk of RPV failure is very low. This should be part of the design basis of the plant.

Originally the design life in the Safety Analysis Report of NPP Borssele was 40 years for the RPV, or 32 full power years with a load factor of 80%. Based on this, fluence calculations and fracture mechanics (Pressurized Thermal Shock) calculations were made to prove the integrity of the RPV for 40 years of operation. To verify the results a surveillance programme was implemented consisting of three sets of specimens comprising of the base metals (two rings), the weld and the Heat Affected Zone of the beltline material. One set (SOP0) was tested unirradiated and the other two sets (SOP1 and SOP2) were irradiated in the RPV and then specimen testing was performed in hot cells to determine the RT_{NDT}. In Figure 32 outside the core barrel ('kernmantel') the 'bestralingskanaal' can be seen where the capsules with specimens are located for irradiation in the RPV.

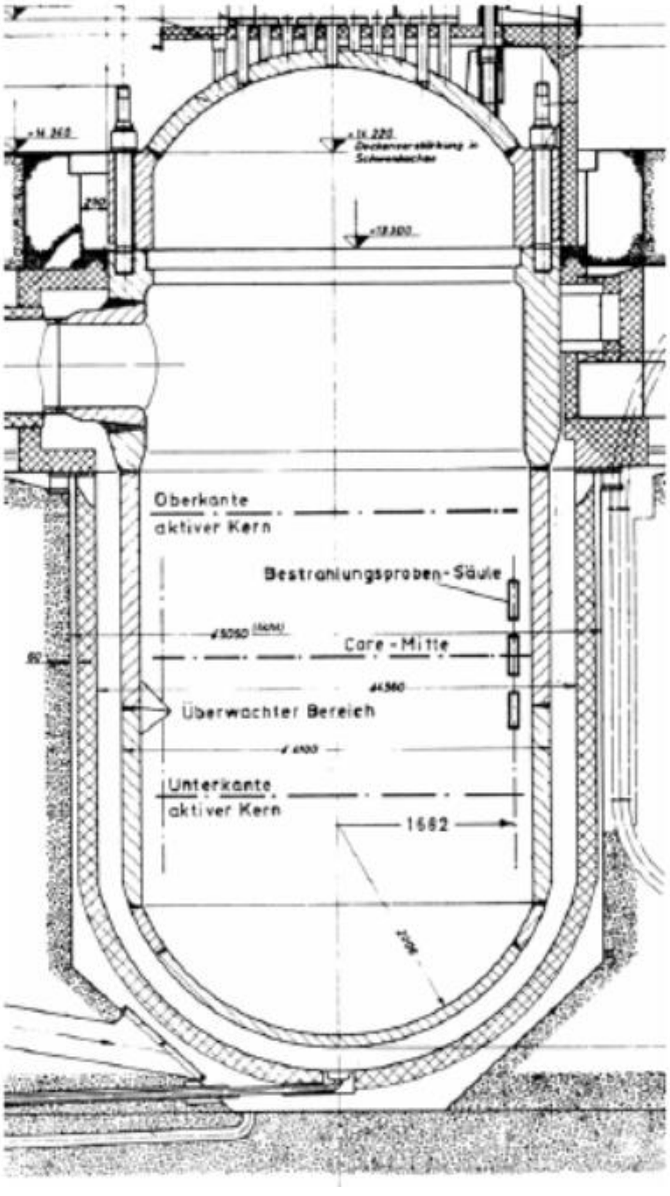

Figure 32 Another view of the RPV of NPP Borssele

Based on both analyses and verification with specimen testing, a conservative RT_{NDT} was determined comprising 40 years of operation. One of the conservative assumptions was to not take credit of the low leakage core which was in place after a few years of operation. This RT_{NDT} was used to set up a pressure-temperature (P-T) limit for safe operation. In 1997 this P-T limit was implemented in the control system of the (new) primary safety valves resulting in a hardware solution against brittle fracture of the RPV by electrical pilot valves with a temperature controlled set point.

To prove safe Long Term Operation (1973 – 2033: 60 years) a state of the art assessment of the RPV was performed regarding embrittlement of the RPV. Also a new surveillance programme was introduced. The following tasks were performed.

- Three new sets of specimens of original material were manufactured (SOP 0a, 3, 4) according to KTA 3203[45].
- SOP3 and 4 were inserted in the RPV in 2007.
- SOP0a was not irradiated. This new unirradiated specimen set was introduced because of a change in the required specimen orientation according to the forenamed KTA 3203 and the introduction of fracture toughness specimens in the surveillance programme.
- The specimen sets comprised both Charpy V (RT_{NDT}) and fracture toughness (RT_{T0}, Master Curve) probes.
- The specimen sets comprised the two base metals and the weld (so-called beltline materials).
- Comprehensive 3D and 2D fluence calculations were made.
- In the fluence calculations the implementation of MOX fuel was also calculated because of the plans to implement MOX. At the plant MOX was introduced in 2014.
- Experimental validation of the fluence calculations by radiochemistry analyses of scraping samples of the austenitic cladding taken in 2010.
- New Thermal Hydraulic calculations as input for the Pressurized Thermal Shock calculations.
- New Pressurized Thermal Shock calculations comprising the complete RPV (also flanges and nozzles outside the highly irradiated areas).

In the analyses 55 effective full power years (EFPY) were assumed meaning operation in the upcoming period to the end of lifetime (2033) with a conservative high load factor of the reactor. With the (experimentally validated) fluence calculations it could be proven that the fluence after 55 EFPY will be below 3,5 10^{19} n/cm^2 which was the design fluence used for the original assessment for 40 years of operation. This under the assumption that the low leakage core configuration will be maintained for the rest of the operating lifetime.

It could be proven that the highest RT_{NDT} was lower than the RT_{NDT} which is used as a basis for the existing pressure – temperature limits at the plant. It could also be proven that for the complete RPV a high margin against brittle fracture is in place for all severe loading conditions.

[45] Safety Standards of the KTA, Surveillance of the Irradiation Behaviour of Reactor Pressure Vessel Materials of LWR Facilities, KTA 3203, Version 6/01

In 2013 after 7 operation cycles specimen set SOP3 was removed from the RPV. SOP3 comprises about 60% of the assessment fluence. The specimens were tested in the hot cells of AREVA Erlangen in 2014.

Adjusted reference temperatures for the assessment fluence are determined using prediction formulas in combination with the results of the surveillance programme. For the RT_{NDT} this has been done by means of Reg. Guide 1.99 Rev.2[46] and IAEA TRS No. 429[47] for the RT_{T0} (based on the fracture mechanics concept). The highest values resulting from this are called RT_{LIMIT}. This value can be compared to the limit curve give in KTA 3203[48]. According to the limit curve an RT_{LIMIT} of 65°C is allowed for a fluence of 3,5 10^{19} n/cm^2. The results of the assessment show that the RT_{LIMIT} of the Borssele RPV will be quite below that limit, see Table 12.

Concept	Limiting material	RT_{limit} [°C]	Difference to KTA 3203 RT_{limit} [K]
RT_{NDT}	BM Ring 03	19	46
RT_{T0}	WM	3	62

Table 12 RT_{LIMIT} of RPV NPP Borssele

It can also be seen that the RT_{T0} concept (Master Curve concept) performed according to IAEA TRS No. 429 results in a lower RT_{LIMIT} than the RT_{NDT} (Charpy-V concept).

Based on these results no extra preventive measures against brittle fracture are necessary for the RPV of NPP Borssele. SOP4 is scheduled to be taken out in 2018. The aim is to take SOP4 out after the specimens have reached 100% of the assessment fluence. In case the low leakage core loading will be changed an assessment will be made to verify if the safety margins for the RPV are still maintained.

Research projects CARISMA, CARINA and CAMERA

EPZ was involved in the related research projects CARISMA[49] and CARINA[50] and is still involved in the ongoing project CAMERA[51]. In these projects highly irradiated materials including materials which are comparable to the materials of NPP Borssele are tested by AREVA and several aspects are studied. In

[46] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99 (TASK ME 305-4), "Radiation Embrittlement of Reactor Vessel Materials", revision 2, May 1988

[47] IAEA Guideline TRS No. 429, Guideline for the Master Curve Approach to Reactor Pressure Vessel Integrity in Nuclear Power Plants, March 2005

[48] Safety Standards of the KTA, Surveillance of the Irradiation Behaviour of Reactor Pressure Vessel Materials of LWR Facilities, KTA 3203, Version 6/01

[49] H. Hein, E. Keim, H. Schnabel, T. Seibert, A. Gundermann: Final Results from the Crack Initiation and Arrest of Irradiated Steel Materials Project on Fracture Mechanical Assessments of Pre-Irradiated RPV Steels Used in German PWR, Journal of ASTM International (2010), STP 1513 on Effects of Radiation on Nuclear Materials and the Nuclear Fuel Cycle: 24th Volume

[50] H. Hein, E. Keim, H. Schnabel, T. Seibert, A. Gundermann: Final Results from the CARINA Project on Crack Initiation and Arrest of Irradiated German RPV Steels for Neutron Fluences in the Upper Bound, Journal of ASTM International (2014), STP 1572 on Effects of Radiation on Nuclear Materials: 26th Volume

[51] H. Hein, J. Barthelmes, C. Eiselt, E. Keim, J. May, F. Obermeier, H. Schendzielorz, H. Schnabel, "Summary and conclusions of the last decade's research on irradiation behaviour of German RPV steel", IGRDM-19, Asheville (NC-USA), 15 – 19 April 2016

the safety assessment for the RPV of NPP Borssele also some relevant results of the CARISMA projects are used to benchmark the results for the RPV.

Ageing under Cyclic or Transient Loading (Fatigue)

All metals that are subject to cyclic mechanical and/or thermal loading are generally susceptible to fatigue, depending on material properties and the presence of inhomogeneity's, design and surface quality, residual stresses, loading conditions and the environment. Fatigue can result in crack initiation and further crack growth potentially leading in to failure of the (sub)component. To mitigate fatigue the application of sufficient design margins and minimization of thermal gradients is necessary. Fatigue management is focused on the prevention of crack initiation.

At NPP Borssele fatigue analyses have been made in the design phase for high safety class primary components and other components. Based on conservative load assumptions (severity and number of transients) it could be proven for safety class 1 and part of safety class 2 mechanical components, that the Cumulative Usage Factor (CUF) will be below 1 meaning that crack initiation will not occur. To check if the loading assumptions are still valid every year the real transients that occurred are counted and compared to the assumed transients.

As part of the project to prove safe Long Term Operation (to 2034, 60 years) a specific project was performed to revalidate the fatigue analyses for the new operation period. The project is described in report NRG-22488/11.106369[52]. It could be proven that also for 60 years of operation the CUFs will be below 1. In this calculation of the CUF also state-of-the-art insight and guidelines on the influence of water (environmental fatigue, see corrosion fatigue) is incorporated.

In the scope of the RPV fatigue analyses are in place for most of the subcomponents. A relative high CUF is only applicable for the RPV studs due to the yearly opening of the vessel head for refuelling and inspections. Besides the yearly monitoring of the transients (including opening of the vessel) and comparison with the assumptions, the RPV studs are inspected for cracking every 10 years according to ASME XI.

Boric Acid Corrosion (BAC)

Boric Acid Corrosion is a form of General Corrosion that attacks Carbon Steel and Low-Alloy Steel in the presence of hot, concentrated aqueous solutions of Boric Acid. It does not generally occur in normal operating or outage conditions within the primary circuit because the components are normally not exposed to borated water. The appearance of Boric Acid Corrosion and the consequences came quite clear in the event of Davis Besse in 2001[53]. This event was assessed at NPP Borssele.

Within the scope of the RPV AMP, Boric Acid Corrosion might occur at the external surfaces of Low-Alloy Steel including mechanical fasteners in case of long periods of leakages from leaking flange connections or through wall cracking.

[52] LTO Demonstration of Fatigue TLAAs, NRG, NRG-22488/11.106369 Revision 1, May 30th 2012
[53] Significant Event Report WANO SER 2002-3, Reactor Pressure Vessel Head Corrosion at Davis Besse, July 2002

Pitting Corrosion

Pitting Corrosion is a form of localized corrosion attack of metallic surfaces in aqueous solution which are passivated by the surrounding environment. Prerequisites for the appearance of Pitting Corrosion often include the presence of a critical concentration of Chloride ions and exceeding a specific electrochemical threshold potential value, designated as the “Pitting potential” of the corrosion system.

Pitting corrosion is mitigated in the primary circuit at NPP Borssele because the main coolant has a sufficiently low corrosion potential (i.e., reducing water chemistry conditions) and a constant pH of 7. For internal surfaces made of Austenitic Stainless Steel, the presence of high Chloride concentrations is mitigated by implementation of the NPP Borssele Water Chemistry Programme. Additionally, the presence of Chlorides from sources, such as gaskets or fitting lubricants, is managed at NPP Borssele through the proceduralized use of Chloride-free chemicals and materials.

A significant amount of Pitting Corrosion of external Austenitic Stainless Steel surfaces would only take place in the presence of external Chloride sources (e.g., from lubricants, adhesive tapes, etc.), not due to the presence of Chlorides in main coolant leakage. External Chloride sources are mitigated at NPP Borssele by the prohibition of Chloride-containing tapes, markings, fluids, etc. since 1982.

To manage BAC at NPP Borssele, at first leakages are avoided as much as possible. A leakage detection system is in place in the reactor building which is able to detect small leakages. Further on walk downs are performed during and after the outages on most accessible components. In case of a found leakage a condition assessment is performed. For safety related components a follow-up plan for corrective actions is made if leakages are found.

Crevice Corrosion

Crevice corrosion is not an independent corrosion mechanism, but the enhancement of a common corrosion mechanism (e.g. general corrosion, pitting corrosion) in special crevice conditions.

Critical conditions that could promote Crevice Corrosion of external Carbon or Low-Alloy Steel surfaces could exist if leakages were to occur continuously over a long period of time. However, the visible effect of this incident is the presence of Boric Acid deposits. When Boric Acid deposits are present, the leading ageing mechanism is Boric Acid Corrosion. Therefore, the occurrence of Crevice Corrosion is covered in this respect by the management of Boric Acid Corrosion.

With respect to Crevice Corrosion of internal surfaces the presence of high Chloride concentrations is generally mitigated by implementation of the NPP Borssele Water Chemistry Programme. However, Austenitic Stainless Steel (i.e., RPV cladding) may become sensitive to Crevice Corrosion when impurities (e.g., Chlorides) accumulate in the crevice during operation. This can lead to an increase in susceptibility to pitting or general attack, when the pH in the crevice changes. The contamination of medium and materials with Chlorides and other impurities is managed at NPP Borssele by the Water Chemistry Programme and the restricted use of chemicals and gaskets with critical impurities.

Primary Water Stress Corrosion Cracking (PWSCC) of Nickel-base Alloys

PWSCC is a type of intergranular SCC which may occur[54] in Nickel-base Alloys in contact with high-temperature main coolant. This corrosion mechanism is thermally activated and exhibits an intergranular crack path. Pre-condition for the occurrence of PWSCC is the simultaneous presence of a susceptible material condition (e.g., microstructure), high tensile stresses due to mechanical loading and/or residual stresses, and a corrosive environment (e.g., high temperature medium).

In PWR primary water conditions, PWSCC may occur in Alloy 600 and weld filler metals of similar composition (i.e., Alloy 82 and 182). The Core Barrel Guide Blocks (item 23 in Figure 31) are made of Alloy 600 and the associated welds are made of Alloy 182/82. These subcomponents are in contact with the main coolant. The assembly weld for the Top Head Nozzles (W16) is also made of a Nickel-base Alloy, but the design ensures that coolant will not come into contact with this Nickel-base subcomponent and PWSCC can, therefore, be excluded for the Top Head Nozzle weld. Therefore PWSCC is only relevant for the CB Guide Blocks (including the relevant welds).

The ageing management for potential PWSCC of the CB Guide Blocks is periodic inspection. However at NPP Borssele before the AMR visual inspection VT-3 was in place for this and it was recognized that VT-3 is not adequate for this. Therefore based on the AMR VT-1 inspection was implemented in the In-Service Inspection programme. The results of this inspection in the outage of 2013 showed no signs of cracking in the CB Guide Blocks and the welds.

Corrosion Fatigue (Environmentally Assisted Fatigue, EAF)

Corrosion Fatigue is a relevant ageing mechanism for all (sub)components subjected to cyclic loading under simultaneous environmental impact. An environmental effect on the Fatigue life of Low-Alloy Steel and Austenitic Steel was shown in experimental LWR coolant environments (NUREG/CR-6909[55]). In the fatigue curves of the original design codes (e.g. ASME III) the influence was not conservatively enough addressed was concluded. Based on that guidelines NUREG/CR-6909 came in place to be able to incorporate the reduction of fatigue life by the water environment. The most used way to do this is to calculate correction factors for the CUF based on several parameters to be determined for the specific location.

At NPP Borssele Corrosion Fatigue was taken into account in the revalidation of the fatigue analyses for 60 years of operations. To do this a (at that time) draft German KTA guideline 3201.2 (Teil 2)[56] was followed. In this guideline screening criteria are used for Corrosion Fatigue, 0.2 for austenitic steel and 0.4 for ferritic steel. If the calculated CUF is below this thresholds than it can be concluded that the effect of the water medium will not reduce the fatigue life. If the CUF is above this, correction factors can be determined according to NUREG/CR-6909. If the resulting CUF is still below one, no crack initiation by fatigue is expected to occur during the assumed operating life.

[54] Stress Corrosion Cracking in Light Water Reactors: Good Practices and Lessons Learned, IAEA Nuclear Energy Series, No. NP-T-3.13, 2011

[55] Chopra, O.K., W.J. Shack, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials, NUREG/CR-6909, November 2006.

[56] KTA 3201.2 Komponenten des Primärkreises von Leichtwasserreaktoren, Teil 2: Auslegung, Konstruktion und Berechnung, Änderungsentwurf, 01-01-2011

For all the RPV subcomponents in contact with water it could be proven[57] that the CUF will be below the thresholds of 0.2 or 0.4 respectively. The highly loaded RPV studs are not in contact with water during fatigue loading.

5.1.4 Monitoring, testing, sampling and inspection activities for RPV

In section 5.1.2 the ageing management of the RPV of NPP Borssele is dealt with from the perspective of potential significant ageing mechanisms. Already several monitoring, testing, sampling and inspection activities are mentioned there to manage the particular ageing mechanism. In the present section the applicable and available generic monitoring, testing and inspection programmes are described which are in place at NPP Borssele and which are relevant for the ageing management of the RPV.

The following programmes are relevant and therefore are briefly described:

- Water chemistry programme
- Fatigue Loads monitoring
- Leakage Monitoring
- Loose Parts Monitoring
- In-Service Inspection Programme
- Surveillance Programme for neutron induced embrittlement

Water chemistry programme

The Water Chemistry Programme at NPP Borssele is an existing programme that monitors and adjusts the chemistry properties in the Primary Systems, Secondary Systems, Nuclear Safety Systems, Safety-Related Auxiliary Systems, and in the Heating, Ventilation and Air-Conditioning Systems. The programme relies on monitoring and control of main coolant water chemistry based primarily on the VGB guidelines for primary and secondary water chemistry, as well as external experiences (e.g., from EPRI, EDF, and WANO).

The primary objectives of the Water Chemistry Programme include:

- Minimization of corrosion and erosion in primary and secondary systems,
- Prevention of fuel element degradation due to corrosion,
- Minimization of the occurrence of deposit accumulation within primary and secondary systems,
- Minimization of increases in radiation levels in the primary systems, and
- Minimization of radiation dose to people and the environment.

Monitored Parameters: The concentration of certain chemical species is monitored to mitigate the degradation of plant SSCs, including Chlorides, Fluorides, Sulfates, dissolved Oxygen and Hydrogen. Water quality (i.e., pH and conductivity) is also monitored to facilitate maintenance within the limits

[57] LTO Demonstration of Fatigue TLAAs, NRG, NRG-22488/11.106369 Revision 1, May 30th 2012

stated within relevant guidance documents. These parameters are monitored through the use of in-process methods or sampling.

Detection: Chemistry programmes allow the detection of deviations from desired water chemistry settings. The use of these programmes does not directly detect the presence of an ageing mechanism in progress. However, One-Time Inspections are periodically performed in representative areas of certain relevant systems to determine the viability of the current water chemistry settings (i.e., that the set limits are indeed mitigating the corrosion of internal surfaces).

Monitoring and Trending: Sampling is performed based on the frequency and direction provided within the VGB guidelines. Increased sampling frequencies may be utilized when adverse conditions (i.e., abnormal chemistry conditions) are experienced. The results of chemical analyses and calibration activities are reported as guided by plant-specific documentation.

Condition Assessment: Any evidence of adverse conditions (i.e., ageing mechanisms in progress or unacceptable water chemistry results) is evaluated to determine the root cause and allow correction of the condition.

Verification of effectiveness: this is performed of both the chemistry control programme and the corrective actions. This is to ensure that no significant amount of degradation is occurring and that the component-intended function can be maintained during operation.

Corrective Actions: When chemistry parameters are reported to be outside the allowable range, actions are taken to return the deviating parameter to concentrations within the acceptable range within the time period specified in plant-specific guidance documents.

Leakage Monitoring

Leakage detection is required to identify critical leakage in a timely manner to ensure that adequate corrective measures can be taken. Leak detection in larger areas is generally aggregate (i.e., accumulation from a selected group of rooms or areas) and not restricted to specific rooms or systems. Leakage monitoring in the scope of ageing management is intended to identify critical leakage in long-lived passive components. Dedicated leakage monitoring systems for active and short-lived passive components such as seals and O-rings are not intended for ageing management purposes and are therefore not addressed in this chapter.

All monitoring systems detect deviations from normal operation.

Detection: Several methods are implemented at NPP Borssele to detect, measure, and locate leakage. Leakage monitoring inside Containment is performed using, for example, the following information:

- Containment air cooler condensate flow collection
    - Monitoring of the moisture content in the air through the condensate level in the condensate vessels within the safety important HVAC systems.
- Radioactivity monitoring
    - Detection of increased radioactivity in the vented air within the safety important HVAC systems, and

    - Main coolant leakage through the steam generator U-tubes to the secondary side, detected by measuring radioactivity in the secondary circuit (main steam system).
- Temperature monitoring within the HVAC system for cooling of the concrete shield around the RPV, and
- Performance of walk downs during leakage tests (i.e., system pressure tests) to identify leakage.

Corrective Actions: Once leakage is detected, actions are taken by the plant to repair relevant equipment. Maintenance or other corrective actions (Non-Destructive Testing) are performed by NPP Borssele depending on the severity of the leakage to prevent failure of a component.

Preventive Actions: ISI and preventive maintenance, such as the replacement of components or subcomponents (i.e., seal parts) and vibration measurement, are performed to take the necessary actions to prevent failure of a component.

Loose Parts Monitoring

The Loose Parts Monitoring System (KÜS – System), is an existing condition monitoring programme installed during the 1984 outage to continuously monitor and detect the presence of loose parts within the Main Coolant System. Potential sources for loose parts include damaged components or foreign materials unintentionally left within a system following an outage (or other maintenance period), as well as from the original manufacturing and installation period. In the event that a loose part is detected (depending on the mass and location of the part, as well as whether it is stationary or moving), corrective actions are initiated to prevent subsequent damage (e.g., impact and wear) from occurring.

Scope: Detection is mainly focused on loose parts within the primary circuit, including the internals of the components that form a part of the Main Coolant System.

Detection: Piezoelectric detectors have been installed on the external surfaces of the Reactor Pressure Vessel and the Steam Generators at fixed strategic locations. Two detectors have been installed on the RPV closure head, one detector has been installed on each inlet to the SG primary side channel head, and one detector has been installed on each SG feedwater nozzle. Loose parts generally collect in these areas, when circulated within the Main Coolant System or secondary circuit.

Condition Assessment: The sound generated by the impact of a loose part is detected by the piezoelectric detectors and relayed to the monitoring system, which is continuously monitoring within the audible or ultrasonic frequency range. An alarm indication occurs if the threshold is exceeded and the impact is subsequently recorded. Routine post-event monitoring is performed periodically by listening to the acoustic signals transmitted through speakers, as the human ear can discriminate signals from background noise.

So far a few times the installed loose parts monitoring system indicated unusual sounds. In these cases the RPV was not involved and the integrity of the RPV was not affected.

In-Service Inspection Programme

In-Service Inspections are performed to assure integrity of structures and components of the NPP. By means of qualified volumetric, visual and/or surface examinations SC are tested.

The ISI-programme at NPP Borssele is based on the ASME XI guidelines and requirements of the Dutch Law for pressure equipment enhanced with inspections based on operating experiences and regulatory requirements. Based on the Ageing Management Review for Long Term Operation also extra inspections were introduced partly one-time and partly added to the programme.

Nowadays NPP Borssele follows a ten-yearly inspection interval meaning the timeframe in which all required inspections have to be performed. The current interval ranges from 2010 until 2019 and is based on ASME XI edition 2007.

Besides the normal scope of the regular ISI-programme the following inspections have been performed for the RPV:

- Based on the experiences with boric acid corrosion at NPP Davis Besse[58] a visual inspection of the RPV head is performed. No signs of corrosion could be found so far. The surface of the RPV head is in a good condition.
- Based on the findings at the Belgium plants Doel-3 and Tihange-2[59] a qualified volumetric inspection was performed in 2013 to assure that the RPV of NPP Borssele is free of hydrogen flakes or other kind of laminar indications. Based on the inspection it can be concluded that the RPV is free of laminar indications.
- In 2016 a qualified one-time volumetric inspection was performed on the ligaments of the RPV head as part of the plant's LTO-project. The problems that were experienced at the Belgian, Swiss and French NPPs were considered when qualifying the inspection, but were not the reason for performing the inspection in itself. No relevant indications have been detected.
- In 2017 a state-of-the-art inspection on the inside of the vessel has been done focused on potential underclad cracks in the vessel wall. Until the late 1980s also periodic inspections focused on potential underclad cracks were done. For LTO it was decided to do a one-time inspection according to state-of-the-art inspection possibilities. No underclad cracks have been found.
- As mentioned earlier based on the outcome of the Ageing Management Review the visual (in-service) inspection of the Core Barrel Guide Blocks has been changed from VT-3 in VT-1. The inspection of 2013 showed no signs of cracking.

All the inspections are performed by certified inspection firms. Based on the Dutch law the nuclear regulator (ANVS) has delegated the surveillance (including qualification) of the approved ISI programme to an independent certified third party.

In case of findings of not fulfilling the acceptance criteria at least a condition assessment has to be performed. Further operation with or without repair has to be justified in a written document and approved by the regulator.

[58] Significant Event Report WANO SER 2002-3, Reactor Pressure Vessel Head Corrosion at Davis Besse, July 2002
[59] roceedings of the 2016 ASME PVP conference (PVP2016-63878) Overview of the Doel 3 and Tihange 2 Reactor Pressure Vessel Safety Cases, M. de Smet, J. van Vyve

Surveillance Programme for neutron induced ageing

As mentioned earlier to monitor neutron induced irradiation embrittlement, a surveillance programme is in place in which specimens of the beltline materials are irradiated in the RPV during operation. The original programme with three sets of specimens was enhanced with three extra sets of specimens because of the extension of the operation lifetime from 40 to 60 years of operation. The last specimen set will probably be taken out in 2018 because then the specimens will have received about 100% of the assessment fluence.

Load monitoring for fatigue management

As mentioned already in section 5.1.2 fatigue loads are monitored and reported every year to check if the occurred loads are still in line with the assumptions of the fatigue analyses. The assumed loads are written in a load catalogue. Besides the studs the fatigue loading of the RPV is pretty low. For the studs in-service inspections are in place to be sure that the integrity is maintained.

5.1.5 Preventive and remedial actions for RPVs

For the management of the RPV of NPP Borssele the following preventive actions are in place.

Water chemistry programme

Besides monitoring as described in section 5.1.3 this programme also is a preventive programme because of the control of the water chemistry. In case of operating outside the chemical parameters a condition assessment of the affected SSC will be done. If needed repair measures will be taken.

P – T limits to avoid brittle fracture of the RPV

Related to the ageing mechanism neutron induced ageing P – T limits are in place to avoid brittle fracture of the RPV. These (conservative) limits are incorporated in the control system of the primary safety valves. If the limits are violated emergency procedures are in place to restore the safe situation.

Prohibition of Chlorides on external surfaces

To avoid corrosion mechanisms like pitting corrosion on external (austenitic) surfaces chloride-containing tapes, markers, fluids, etc. are prohibited since 1982. An administrative procedure is in place combined with a database with allowed and not-allowed materials.

5.2 Licensee's experience of the application of AMPs for RPVs

From the ageing mechanisms applicable for the RPV as described in section 5.1 (NPP sections), neutron induced ageing is the most important mechanism and the only one which can't be prevented completely. Particularly for this mechanism very conservative assumptions were made in the design phase and during design conservative limits were set up to be very sure that brittle fracture will not occur. Particularly in the comprehensive safety assessment made to prove safe long term operation it could be proven that the ageing (embrittlement) is far less than assumed in the design phase. This combined with comprehensive volumetric in-service inspections shows that very high safety margins are available for the RPV of NPP Borssele.

External incidents like severe corrosion of the RPV head at Davis Besse and the finding of hydrogen flakes in the RPVs Doel-3 and Tihange-2 were thoroughly assessed and led to specific inspections to prove that these issues are not applicable for the RPV of NPP Borssele. In all of these inspections it was confirmed that those issues are not applicable for the RPV of Borssele.

It can be concluded that the current AMP of the RPV of NPP Borssele is adequate. However the AMP and the results will be evaluated periodically and if needed changes of the AMP will be implemented. In this respect experience feedback including experiences from other NPPs is very important. Possible relevant information about ageing of the RPV will be reviewed to look for gaps and need of improvement of the AMP of the RPV. Although fundamental R&D was done in the past and comprehensive insight is available on the RPV, EPZ will try to follow as much as possible the current state-on-the-art in this field and will act upon this if relevant results come out.

5.3 Regulator's assessment and conclusions on ageing management of RPVs

The most important ageing issue is the potential embrittlement of the material. During the LTO-programme again a complete ageing management review of all aspects was done, including again introduction in 2007 of test specimens in the available irradiation chambers. With the conservative assumption of 55 FPY produced electricity at 60 years of the life the calculated fluency is below the assumed value for the design in 1973. Taking into account the first specimen assessments after 7 years (2014) at 60% of the 55 FPY, the RT-limit values are well below the norm, confirming a large margin. In 2018 the specimens have reached 100% and again the effects will be assessed. Based on foreign experiences like Davis Besse, the hydrogen flakes in Doel/Tihange and the Carbon Segregation issue additional inspections/measurements were done, that showed that these issues are not impacting the RPV. Also recently, a modernized measurement was done to detect underclad cracks. None were found.

ANVS concludes that this AMP is effective.

Further refer to section 2.7.

6 Calandria/pressure tubes (CANDU)

Not applicable.

7 Concrete containment structures

This chapter only applies to the Borssele NPP.

7.1 Description of ageing management programmes for concrete structures

7.1.1 Scope of ageing management for concrete structures

The primary containment of the Borssele NPP consists of a steel sphere, designed to withstand the pressure associated with a significant leakage of coolant from the reactor cooling system (Loss of Coolant Accident – LOCA).

An ageing management review was conducted on the steel containment sphere in preparation of safe long term operation (LTO), and it has its own dedicated ageing management section in the plant's living ageing management process[60]. However, steel containment structures are not described in the technical specifications for the scope of this topical peer review and will therefore not be discussed further in this report. The scope of this chapter will therefore focus on the concrete structure that surrounds the steel primary containment sphere.

The Reactor Annulus Building (Building 02) has an outside diameter of 49.5 m, while the steel containment sphere has a diameter of 46.0 m. The lower section of reinforced concrete containment (secondary protection) of the reactor annulus building, with a minimum wall thickness of 0.6 m, is constructed as a cylinder up to a height of 32.5 above sea level, and the upper section is dome-shaped, see Figure 33.

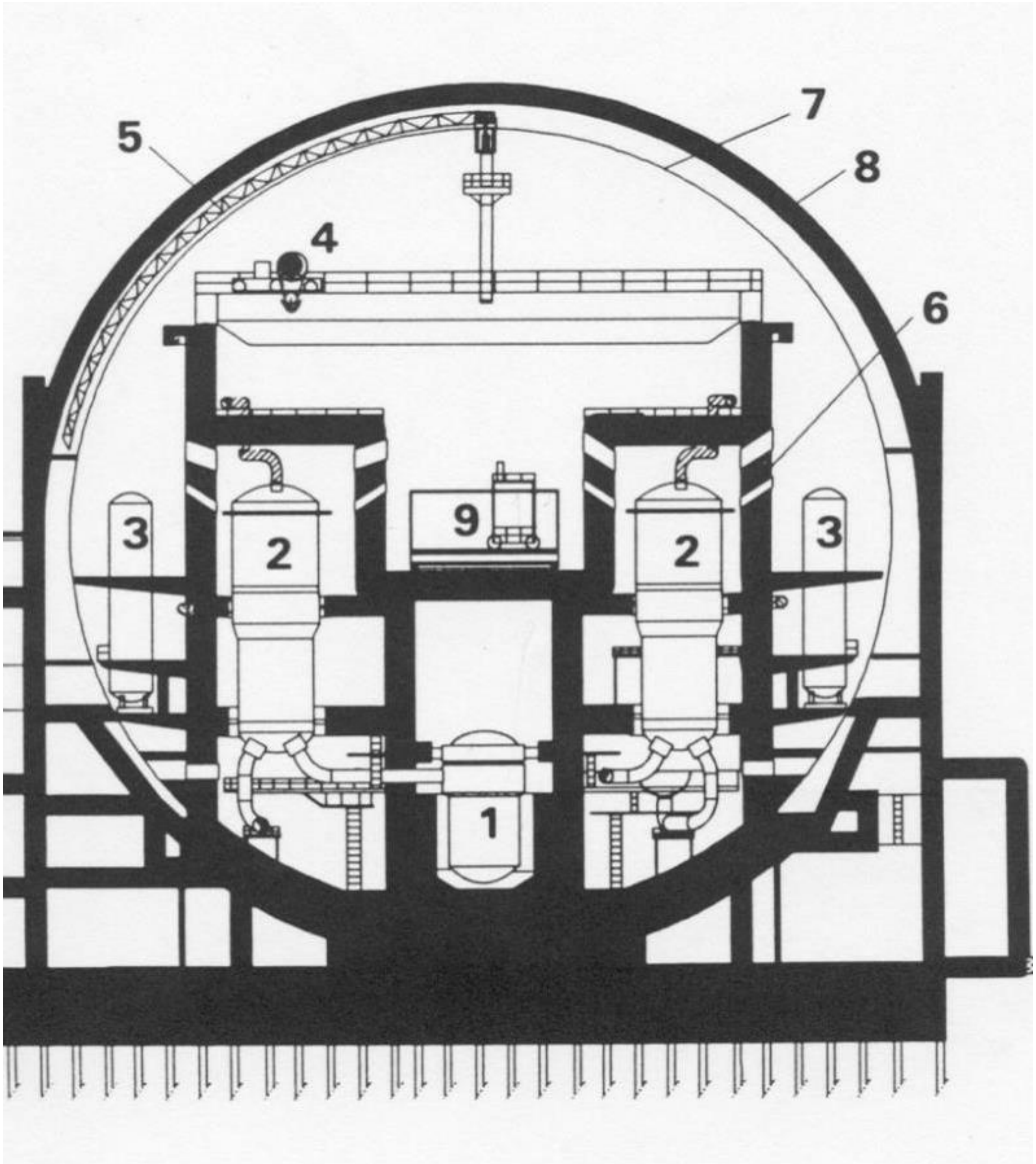

Figure 33 Reactor annulus building (Building 02), Borssele NPP

[60] PU-N12-50, Ageing management process

The foundation of Building 02 comprises two reinforced concrete foundation plates laid on top of each other, supported by piles. A pressure-water tight layer of bitumen/foil-insulation is fitted between the upper and lower foundation plates, and extends to a height of 5.0 m above sea level on the sides of the building.

The wall and floor thicknesses were designed in accordance with the requirements for radiation protection. The interior walls and floors of the annulus are made of reinforced concrete. The lower section of the primary containment sphere is embedded in a reinforced concrete bowl. The loads of the structures inside the containment sphere are discharged via this bowl.

A decontaminable coating is applied to the surfaces of the walls and ceilings in the annulus, which is resistant to the ambient conditions during LOCA conditions (steam at 130ºC). A layer of epoxy resin has been applied to the floors of these areas.

The primary containment protects the environment from the release of radioactive substances. The purpose of the secondary dome is to protect the primary steel sphere from external impacts due to earthquake, explosion pressure waves, private aircraft crashes or external flooding.

7.1.2 Ageing assessment of concrete structures

Ageing mechanisms that are relevant to the Reactor Annulus Building (Building 02) and that require ageing management activities have been established in accordance with the method as described in chapter 2 of this NAR on the overall ageing management programme requirements and implementation under the guidance of the CAM-SC[61]. Relevant ageing mechanisms have been identified after careful consideration of the operating conditions, material properties and environmental conditions of the Reactor Annulus Building. These relevant mechanisms are described in Table 13. As is the case for the CAM-MC, experts from the OEM compiled the Catalogue of Ageing Mechanisms for Structural Components (CAM-SC), containing the latest insights on the subject of structural component ageing.

In the case of buildings, Table 13 lists the relevant SSC or part thereof as commodity group, not as individual system element. The table further lists the material the commodity is constructed from, and describes the environmental conditions. Based on this information, relevant ageing effects were identified, and the ageing mechanisms that would be responsible for these ageing effects. The last column of the table lists the ageing management activities to manage the relevant ageing mechanisms.

[61] PU-N12-50-103, Catalogue of ageing mechanisms, structural components

Table 13 Ageing Measures for the Reactor Annulus Building (02)

SSC	Material	Environment	Ageing Effect	Ageing Mechanism	Ageing Measure at KCB
Concrete Roof	Reinforced Concrete	• Air – Outdoor • Raw Water – Flowing	Loss of Material	• Freeze – Thaw • Corrosion of Embedded Steel and Steel Reinforcement • Reaction with Aggregates	**Roof** • **General inspection of building roofs, incorporating:** → Visual inspection of existing concrete structures → Inspection of concrete conservation measures of floors, walls, ceilings and constructions • **Inspection of building roofs on specific requirements** → Using the visual inspection of existing concrete structures as a work instruction
			Cracking	• Freeze – Thaw • Reaction with Aggregates • Fatigue	
			Change in Material Properties	• Leaching of Calcium Hydroxide • Carbonation • Aggressive Chemicals	
Concrete Wall/ Foundation	Reinforced Concrete	• Air – Indoor • Air – Outdoor • Air with Borated Water Leakage • Raw Water – Stagnant • Raw Water – Flowing • Soil/Groundwater • Buried	Loss of Material	• Freeze – Thaw • Erosion, Abrasion or Cavitation • Elevated Temperature • Corrosion of Embedded Steel and Steel Reinforcement	**Walls** • **Inspection of building facades** • **Inspection of facades on specific requirements** → Using the visual inspection of existing concrete structures as a work instruction **Indoor** • **Inspection of building walls, incorporating:** → Visual inspection of existing concrete structures → Further investigation in damage

SSC	Material	Environment	Ageing Effect	Ageing Mechanism	Ageing Measure at KCB
			Cracking	• Freeze – Thaw • Reaction with Aggregates • Elevated Temperature • Fatigue	and hidden deficiencies of concrete structures → Root cause analysis of damage to concrete structures and prognosis of future damage • **Inspection of walls on specific requirements**
SSC	**Material**	**Environment**	**Ageing Effect**	**Ageing Mechanism**	**Ageing Measure at KCB**
Concrete wall / foundation	Reinforced concrete	• Air – Indoor • Air – Outdoor • Air with Borated Water Leakage • Raw Water – Stagnant • Raw Water – Flowing • Soil/Groundwater Buried	Change in Material Properties	• Leaching of Calcium Hydroxide • Carbonation • Elevated Temperature	**Foundations** • **Inspection of building foundations, incorporating:** → Visual inspection of existing concrete structures → Further investigation in damage and hidden deficiencies of concrete structures → Root cause analysis of damage to concrete structures and prognosis of future damage • **Inspection of foundations on specific requirements** (Seismic) • **Settlement survey for buildings**
			Loss of Material / Change in Material Properties	• Aggressive Chemicals	**Foundations** • **Periodic groundwater analysis in the vicinity of the foundations**

7.1.3 Monitoring, testing, sampling and inspection activities for concrete structures

The Reactor Annulus Building (02) is one of the concrete buildings that house nuclear safety relevant systems. These buildings are subject to SSC-based commodity group inspection programmes and specific civil maintenance programmes. Individual inspection programmes contain a collection of underlying maintenance procedures. Specific requirements are complied with, by using qualified external inspectors. The settlement survey, for example, is conducted in cooperation with the association of Dutch companies in geodesy and geo-information “Vereniging van Nederlandse Bedrijven in de Geodesie en Geo-Informatie” (VNBG).

Inspection frequencies are set according to the requirements of the respective ageing mechanism and the safety relevance of the system. The applied frequency of these repetitive inspections should impede further deterioration and if applicable, prevent transition to adjoining components.

Furthermore, the ageing management programmes comply with the level of personnel requirements that are imposed by the underlying standards, and keep to the acceptance criteria that these standards adhere to.

It can be seen from Table 13 that all relevant ageing mechanisms for the Reactor Annulus Building are conducted in a number of ageing management programmes. The ageing management programmes that are relevant for the Reactor Annulus Building are summarised in Table 14 and explained below.

Table 14 Ageing Measures for the Reactor Annulus Building (02)

Name of programme	Inspection interval
Inspection of building foundations, including specific requirements	6 years (general) 3 years (seismic requirements)
Inspection of concrete indoor walls, including specific requirements	5 years (confinement, flooding and radiation protection)
Inspection of concrete outdoor walls (facades), including specific requirements	1 year (fire protection) 5 years (confinement, flooding and radiation protection) 10 years (explosion wave resistance)
Inspection of ceilings, including specific inspection requirements	1 year (fire protection) 5 years (confinement, flooding and radiation protection)
Inspection of building roofs, including specific requirements	1 year (fire protection) 3 years (seismic requirements) 5 years (radiation protection) 10 years (explosion wave resistance and external impact)
Settlement survey of concrete buildings	5 years

<u>Ageing management of building foundations, including specific requirements</u>

The ageing effects Loss of Material, Cracking and Change in Material Properties are examined by the inspection programme on building foundations, including its underlying maintenance programmes.

This inspection programme is based on normal maintenance inspection activities of the following sub-programmes:

- Visual inspection of concrete structures;
- Settlement survey of concrete buildings;
- Further investigation of damage and hidden deficiencies of concrete structures;
- Investigation into the damage to concrete structures, its cause, and prognosis of future degradation.

The inspection programme for the specific requirements is directed at the requirements for a seismic class I building. Included in the scope of the inspections are the foundation mat of the Reactor Annulus Building (02) and the foundation of the secondary containment wall (02).

Inspections also include investigation of any deviation from the design and intended function requirements. The structures are inspected for cracking as part of the maintenance related visual inspection. It is important that there is no cracking of any kind in the concrete or around anchors and their functional elements.

Ageing management of indoor concrete building walls, including specific requirements

The ageing effects Loss of Material, Cracking and Change in Material Properties for Indoor exposed concrete walls are covered by this inspection programme and its underlying maintenance programmes.

Concrete walls experience specific deterioration and are visually examined for cracks. Their coating is to remain compliant with their requirements.

If the inspection results give reason for corrective action or more precise investigations on concrete walls, the remedial maintenance programme activities for examination to find the cause of damage in concrete structures and establishing a forecast of future deterioration are applied. Records are always kept of the inspections, so that trending can be applied in these cases.

Walls in the buildings designed with different intended functions are subject to various inspection requirements for the intended safety functions, i.e., fire resistance, impermeability to fluids, chemical resistance, LOCA resistance, ease of decontamination, seismic resistance, radiation protection, internal flood resistance, external flood resistance, gas tightness and other specific requirements. Structural members that are subject to inspection are individually examined regarding their specific requirements according to their intended function. Any deviation found is recorded in the respective inspection documentation. Remedial actions are launched if the level of degradation exceeds the acceptance criteria of the inspection programme.

Specific fire resistance requires fire retardant materials for wall constructions and structures according to the actual design and any deterioration must still comply with the requirements that were assumed during the original design.

Fluid resistant wall coatings must be fluid-tight and are checked to be undamaged. Alterations from the design are recorded.

Chemical resistant walls and their coatings are checked regarding any attack and their full integrity. Slight discolouration is acceptable whereas blistering; separation, softening etc. are not acceptable.

Signs of saponification are checked in this inspection but not considered as a significant ageing mechanism.

One important inspection is the LOCA resistant coating, which has to be intact and without any damages, and its bond is intact. These attributes are specifically checked and examined visually.

Coating that needs to be easily decontaminated is checked regarding their full integrity and the securing of decontamination, for damage, porosity and wear. Marks of wear or separation require further actions.

Inspection for seismic resistance is executed to confirm the absence of cracks in concrete structures. These are examined and the anchors and fixations are inspected on their appropriate condition. Typically, these inspections will be performed during a refuelling outage, so that accessibility with regard to scaffolding requirements, removal of insulation, etc. can be attended to.

As a subsequent inspection, the resistance against flooding is examined. SSCs must meet the requirements of the original design specifications.

Ageing management of outdoor concrete building walls (facades), including specific requirements

The ageing effects Loss of Material, Cracking and Change in Material Properties for Outdoor exposed concrete walls are covered in this inspection programme on facades and their underlying maintenance programmes.

The underlying maintenance programmes require concrete inspections of exposed concrete. The inspection of the coating is performed for concrete floors and walls. Specific requirement on seismic resistance is concrete integrity as designed, which is checked in the visual inspection programme. Therefore cracking is the main focus to be checked. Cracks around anchors are unacceptable, which means that the trending records in this case consist of the repair and replace activity that resulted if cracks were identified during the inspection. Anchors must be in proper condition. Requirements on inspection for protection against external flood incidences are included in this programme.

The following specific requirements may apply:

- Radiation protection measures require the detailed inspection of structural features according to the actual design.
- Explosion resistance requires the detailed inspection of structural features according to the actual design.
- Requirements for external flooding resistance are checked.
- Requirements on structural resistance against explosion wave and air plane crash require compliance to the actual design and a detailed check for absence of cracking.

In all these cases, it is evident that the results of any deterioration must still comply with the requirements that were assumed during the original design.

Ageing management of roof constructions for buildings, including specific requirements

The ageing effects Loss of Material, Cracking and Change in Material Properties that are applicable to the concrete roof are examined by visual inspections and their underlying maintenance programmes for visual inspection and coatings.

For roof constructions made of concrete the visual inspection is examined according to the requirements for concrete structures. The inspection for roof coating is made simultaneously.

Roof constructions with specific requirements on seismic resistance are checked for degradations which can adversely affect the design and their functionality. A visual inspection checking for cracking is performed according to the afore mentioned visual inspection programme for concrete. The concrete is checked for absence of cracks, which especially around anchors are unacceptable. The anchors must be in good condition.

Resistance against external explosion hazards or against building collapse in case of an aircraft crash requires structural strength in roof constructions for Building 02. Therefore the detailed inspection of structural features according to the actual design and any deterioration must show that the requirements that were assumed during the original design are still complied with.

7.1.4 Preventive and remedial actions for concrete structures

The ageing management procedures indicated above, rely on the implementation of maintenance procedures that the buildings are subjected to for preventive and remedial actions. A description of these activities is provided below.

Table 15 Existing maintenance activities for the Reactor Annulus Building (02), NPP Borssele

Name of inspection	Inspection interval
Inspection of Coatings for Concrete Floors, Walls and Ceilings	Ongoing
Visual inspection of concrete structures	5 yearly
Hidden defects in concrete structures	Consequential to other events
Examination of the cause of the degradation of concrete structures and the forecast of future degradation	As identified

Visual inspection of concrete structures

This inspection focuses on the particular aspects of the SSC-based inspection programmes as described above. Visual examinations are performed, considering the ageing effects Loss of Material, Cracking and Change of Material Properties. The main subjects of the inspections include cracking, scabbing/flaking, as well as assessment of the reinforcement, surface degradation and any remaining damage.

The location, pattern and direction, gradient and depth of any cracking are of significance. It is checked if the cracking arises from the reinforcement bars and if there are any visible indications of rust. It is determined if cracks contain water or are “bleeding”. Cracks are mapped with a crack width ruler and a precision scale crack magnifier. It is determined if the crack shoulders are eroded, and the crack length and centre distance between global parallel cracks is also measured.

Flaking is handled by mapping the location, dimensions, thickness and reinforcement involved. Dissimilar material, such as repair mortar, is identified, which may have caused degradation.

In the case that any reinforcement is exposed, the locations and the direction of the affected reinforcement bars are mapped. The design function of the affected reinforcement bars and surface condition are recorded, as well as the bar diameter. Visible corrosion is investigated more precisely to determine the dimension of corrosion, pitting corrosion, corrosion stains in the concrete or on reinforcement bar surface only, and percentile loss of bar diameter. Indications of concrete surface degradation, such as calcium leaching, carbonation, salt reaction, erosion, pollution, algae, moss and mould are checked consistently, since these phenomena promote the initiation of other ageing mechanisms.

In the event that the indicated damage symptoms are not mentioned above, their nature, location and characteristics are described separately in the inspection report.

Maintenance inspection and remedial actions for coatings of concrete floors, walls and ceilings

The general condition of coatings is reviewed in an initial inspection. Any contamination and fouling is recorded to the inspection reports. Inspection frequencies depend on the classified functions of the coatings, as identified in the maintenance justification reports.

If a failure of a coating is detected during an inspection, the coating system must be repaired by a qualified coating specialist, using qualified coating and a qualified method. If the failure of the coating is not a normal case of wear and tear, reconsidering and redesigning may be necessary.

Acceptance criteria are set according to accepted international standards for blistering and cracking of the coating[62]. Relevant standards that are adhered to in this regard are contained in ASTM[63] and the ISO 4628 suite of standards[64] for concrete coating inspections.

In the inspection records, cracking is distinguished between random pattern and particular pattern. The crack dimensions are characterized in five groups.

Delamination of the coating is classified. The affected layer and the delaminating pattern are the main subject of this part of the inspection. The dimension or degree of delamination is given by percentage.

Chalking of the coating is classified using adhesion tape. The residue is checked and compared with the samples in the acceptance criteria.

Further assessment of damage and hidden defects in concrete structures

The concrete surface is inspected for flaws using a chipping hammer. Flakes of loose concrete debris or a released reinforcement cover that came loose while using the chipping hammer, will be

[62] ACI 349.3R-96, Evaluation of Existing Nuclear Safety-Related Concrete Structures, American Concrete Institute.
[63] ASTM D 3359-02, hechtingstest
[64] ISO 4628/2 ‘blaasvorming’; ISO 4628/4, ‘scheurvorming’; ISO 4628/5, ‘bladders’; ISO 4628/6, ‘verkrijting’

examined. When this examination provides evidence of the presence of any hidden concrete defects, a follow-up inspection will be initiated. The existence of further honeycombs and voids, which would be manufacturing defects that were not identified earlier in the life of the plant, may be identified as well using this inspection method.

If reinforcement bars are exposed, the thickness of the concrete cover is measured using a measuring gauge. If the rebar is not exposed, the thickness of the concrete cover is determined using an electromagnetic device. An average value is calculated from results of measurement in at least six different locations. The probe provides information about the location, diameter of the rebar, individual measurement results and the direction of the rebar.

The depth of carbonation is checked using a concrete chunk that is chipped away from the concrete surface. At the fresh concrete surface, the depth of carbonation is determined with a phenolphthalein spray. Non-carbonated concrete surfaces become purple in colour and damaged concrete remains unchanged. The inspection is performed at six individual locations with at least two measurements in each location. For further actions and assessment, the following are recorded: location of the measurements (defects), depth of maximum carbonation at that location, and the dimension of the measured location.

Determination of the Chloride content in the concrete is identified using concrete samples in the laboratory. The location inspected must be equivalent or close to the position of measurement of concrete cover. The examination is performed in three steps, with continuous depth, to determine the depth of carbonation horizon in the event that deterioration has occurred. Specific attention is given to the level of reinforcement. The evaluation of Chloride content is significant at this level and is recorded in the inspection report. The identification mark of the measured locations, the measured depth and related concentration of Chlorides are necessary in the assessment of the condition of the concrete structure.

To assess the condition of the concrete, sampling is performed in two of the damaged locations and in one non-damaged location within the material. The recorded information shows that the sensitivity of the concrete that has been examined using existing measures to determine degradation due to ageing in comparison with known undamaged locations.

Any mechanical damage of the coatings is also reported in the inspection record. Size and nature (maybe cause) are mentioned, as well as the damage depth (surface only).

Examination of coating adhesion/bond is performed with reference to the required temperature range. The temperature value is recorded in the inspection reports.

Instructions to perform the inspection work for coatings on concrete floors, walls, ceilings and structures can be found in a dedicated maintenance programme.

Ageing management programme for the examination of damage, cause and forecast of future degradation

This programme describes the process of examination of the corrosion of the reinforcement, The reinforcement bars coverage, carbonation depth, and the Chloride content at the reinforcement layer.

Cracking due to strain interference, overload and settlement is dependent on design loads, values of design deformation/deflection, disposition and the structural model.

Microscopic examination shows cracking due to an alkali-silica reaction and concrete reactive aggregates.

Cracking and delamination due to the freezing of voids and cavities are checked and inspected according to the specific requirements of the structure and its intended function.

Freeze/thaw cycles may degrade the microstructure of concrete. Any damage is evaluated with the exposure class and the intended function of the inspected structure.

Thaw agents grow their volume in presence of water. They are normally in solution and, therefore, trapped in voids and cavities in the concrete structure surfaces. Any remarks regarding surface degradation are assessed and further actions are launched, as necessary.

Another effect is mechanical damage of the concrete surface due to dissolved cement paste.

Volume growth of hardened cement paste, caused by reaction with aggregates, may initiate a damage of the concrete structure.

An optional testing method is the microscopic examination of concrete cores or blocks. The microstructure is examined by polarized and fluorescent microscopy. If other inspections do not provide give sufficient evidence of deterioration, these examinations are performed. Specific attention is given to the condition of the microstructure, with respect to crazing, hydration products, capillary porosity, and the presence of fine aggregates or damage of the hardened cement paste. To assess the condition of the concrete, sampling is performed in two of the damaged locations and one non-damaged location within the material. The recorded information shows that the sensitivity of the concrete has been examined using existing measures to mitigate degradation due to ageing.

The moisture content of the concrete is examined, identification of the cement type is performed and the cement content is determined with regard to its ingredients.

Potential measures are performed on the reinforcement steel, indicating hidden corrosion due to shifted potential values. Differing values indicate rebar corrosion and result in further investigation.

Measurement of the corrosion rate is performed by applying a small current with an auxiliary electrode to the surface of the reinforcement and measuring the potential change with a reference electrode on the surface.

Furthermore, the following examinations may be performed in the scope of examination of damage, cause and forecast of future degradation:

- The electrical resistivity of the concrete;
- the compressive strength (examined by non-destructive testing methods using the Schmidt Hammer);
- porosity of the concrete;
- depth of water impression; and
- compressive strength of bore cores.

Forecast of future degradation in this context was traditionally intended to mean to apply engineering judgment in order to be able to provide input into short- and long-term financial investment requirements for LTO of the plant. Since ageing management became an integrated aspect in the maintenance procedures, recordings of these parameters are becoming more and more available to perform trending on them.

7.2 Licensee's experience of the application of AMPs for concrete structures

As is the case for the ageing management of the concealed piping, the assessment in this chapter is based primarily on engineering judgment through the comparison of ageing mechanisms and existing ageing management activities at the plant.

KCB by now has almost ten years of experience in running the programmes. The different inspections, set with the help of external experts, consider the operating experience of other plants and the safety requirements for relevant components.

As an example of the success of the civil maintenance procedures, the plant has successfully addressed ageing-related problems with the coating on several floor surfaces in the nuclear buildings. Coating has been determined to be at the end of its life. Crazing, delamination and flaking of the applied coating indicated that in the event of a LOCA, the coating strength would not have resisted the heat impact. Although this safety function is rather applicable to the inside of the primary containment, the specifications for the floor coatings in the annulus building are similar.

Root cause analysis determined that the following circumstances may have been responsible:

- Inappropriate protection against coastal/weathering impacts and
- The harm of sulphate compounds during construction, which led to a loss of strength of surface concrete. As a result concrete strength was not as designed to perform as a base for the applied coating product.

To eliminate the possibility that coating would fail to withstand the level of heat during and following a LOCA, repair works had been conducted, to replace the coating and the surface concrete, as detected during inspection. At locations where increased degradation of coating due to irradiation may be expected, thin stainless steel sheets are applied.

Another example of the successful application of the civil ageing management programmes is where degradation was identified on the exterior surface of the reactor annulus building during the 5-yearly visual inspection of concrete structures. Hidden defects were discovered upon subsequent investigation, when sections of the concrete surface were removed by chipping. The examination into the cause of this degradation of the concrete surface found chloride ingress into the concrete surface to be the degradation mechanism. Chloride ingress would have been caused by exposure of the concrete to the maritime air environment at the location of the plant. It was determined that the concrete cover of the exterior would be insufficient to keep protecting the reinforcement bars (rebar) of the concrete structure up to and beyond the planned period of Long-Term Operation. It was therefore decided to conduct a project to replace the outside cover of the reactor annulus building, which was successfully completed before entering LTO.

The existing inspection programmes are determined as sufficient and adequately managed so that the intended function(s) will be maintained consistent with the KCB licensing basis for a service period of 60 years, i.e., Long-Term Operation.

7.3 Regulator's assessment and conclusions on ageing management of concrete structures

The purpose of the concrete dome surrounding of the primary steel containment is to protect the primary steel sphere from external impacts due to earthquake, explosion pressure waves, private aircraft crashes or external flooding. The ageing management is based on a list of ageing mechanisms compiled by experts of the OEM, containing the latest insights. A concrete inspection programme is available for the foundation, inner and outer walls, roofs, floors as well as coatings. E.g. flaws in concrete are inspected visually and using a chipping hammer. If needed further analysis is carried out. A whole set of analyses is described. If needed repairs are carried out. For instance in several places the floor coatings were considered at the end of their life and were replaced. Also a concrete repair programme of the exterior part of the reactor annulus building is carried out.

Also this part of ageing management has been scrutinized during the LTO-project. It is considered effective.

Further refer to 2.7.

8 Pre-stressed concrete pressure vessels (AGR)

Not applicable.

9 Overall assessment and general conclusions

9.1 Overall assessment and general conclusions -Matters common to all installations

In the Netherlands one Nuclear Power Plant (KCB) and two Research Reactors (HFR and HOR) participated in the Topical Peer Review Ageing Management.

Regulation of ageing management

Regulation of nuclear installations from the early years till about 20 years ago was mainly focused on Nuclear Power Plants (at that time NPP Borssele and NPP Dodewaard), keeping in mind that regulatory capacity was smaller than to today. Introduction of ageing management therefore was stimulated first in the NPP (around 1990) and later on at the research reactors, first at the largest one (HFR around 2000) and about 10 years later at the HOR. It was also part of PSR and with using IAEA standards as references.

The regulatory requirement to have an ageing management programme was introduced in licences of the NPP and the HFR (roadmap required), but not yet of the HOR. At the HOR this became mandatory from the PSR implementation plan in 2012. In the near future a complete revision of the HFR and HOR licence will include the requirement for an AMP according to SSG-10. The development by IAEA of the SCO-mission (LTO for RR) was a common request by FANC and ANVS and might be considered as a good practice.

In the licence of the NPP there is also a number of NVRs (modified IAEA Standards) related to AM, but they have to be replaced by modern NVRs.

The approach to have NVRs in the licence as a provision has not yet been applied at the research reactors, but is part of the modernisation project of NVRs, that has started in the second half of 2017 and is planned to be finished in 2020. This also fulfils a number of recommendations from the 2014 IRRS-mission to the Netherlands, which will be followed up in the last quarter of 2018.

Supervision of the introduction and improvement of the AMPs

Supervision of the introduction and improvement of the AMPs at the licensees has been done in different ways. Inspections and assessments of first introductions were part of it. Further improvement came from PSRs and IAEA-missions. ANVS verifies the progress and completion of the recommendations. Also IAEA FU missions play a role. Now the LTO-issues for the NPP are coming to an end, the ANVS will develop a new approach to systematically and regularly inspect the AMP. The same applies to the HOR.

Since the HFR's last step of implementation of a modernized integrated AMP is not completed and thus in an earlier stage of development than those of the other two reactors, in the next 2-3 years the ANVS will devote most AMP-supervision effort to the HFR AMP. After a recent inspection ANVS has decided to request from the licensee a detailed implementation plan with milestones and periodic reporting to assure that NRG will sufficiently finish the implementation of the AMP before the IAEA SCO mission. ANVS will closely watch this and monitor if NRG will devote enough manpower and give priority to this activity.

During 2017 the ANVS has discussed the future policy for LTO of research reactors related to the periodic safety reviews. The main steps, apart from the inspection programme, are (1) the confirmation of the SCO (LTO) mission at the HFR, (2) the decision based on a graded approach not to have a SCO mission at the HOR but instead concentrating on the INSARR missions once every 10 years with special emphasis on AM.

9.2 Overall assessment and general conclusions - NPP Borssele

The NPP participated fully in the exercise, including the four components. As required in the TPR specification, the concrete containment was handled differently as described in chapter 7 of the specification:

"*the concrete structure that surrounds a (self-standing) steel containment designed to withstand the pressure associated with a significant leakage from the reactor cooling system. This structure is often the outer wall of the reactor building*"

Originally, many activities now executed under ageing management, were carried out in maintenance programmes. The overall aging management programme has been built up from the first half of the 1990s, when the plant was reaching 20 years of operations, more than 25 years ago. Its maturation since then was further stimulated through Periodic Safety Reviews that were using actual IAEA standards and by inviting some IAEA-missions in 2003 (AMAT) and later on the SALTO-missions during the LTO-project in 2009-2014. Also other sources of information were drivers for further development like VGB, IGALL. The overall AMP is well and in detail described in the integrated management system and a well-defined AM-team coordinates its execution. The LTO results have been assessed by the regulator, with the help of the TSO (GRS). A licence modification for LTO was granted.

The licensee is very actively participating in international activities around AM/LTO. The phase-out of Germany makes it necessary to gradually replace the German source of information. The plant has stated it will keep the high level of international connection in the future e.g. with other countries having KWU-design plants. On request of the ANVS the plant has agreed to start participation in the OECD-NEA CODAP project, as of 2018. The strong international engagement of the NPP could be considered as a good practice.

Currently and in the next few years the remaining actions are carried out related to the LTO programme.

The ANVS considers the overall AMP and its parts dealing with the SSCs as state of the art and complying with international standards.

Electrical cables AMP is structured as all other AMPs. Cables relevant for LOCA or HELB conditions are handled in a special cooperation programme under VGB. Outcome of the LTO-programme was the replacement of some electrical equipment, including cables. A lot of international cooperation is going on with the other KWU plants. A cable deposit is used in one of the German plants. Another solution will be necessary after the German phase-out.

The other (non-LOCA) cables and wires underwent a structures 10-step approach to determine the cables and wires critical to ageing degradation, taking into account all possible influences. It turned out that only about 200 cables were critical. For wires no critical ones were found. Visual and other inspections and test programmes are focused on these cables. Visual inspections also take non-safety related cables for additional experience. Test programmes are aimed at determining and following the actual amount and development of degradation. The environmental conditions, that were the basis of the assessment of critical cables are periodically monitored to determine significant change.

The amount of safety relevant concealed piping is not very high. These systems are the auxiliary and emergency cooling water system (system VF), the back-up residual heat removal water cooling system (System VE) and the low pressure fire extinction system (system UJ), that also has some functions in (severe) accident management. No piping of diesel fuel or radioactive water are concealed. After replacement of relevant parts of VF due to conceptual ageing in 2012 the ageing prevention is delivered by the design, confirmed by inspection programme. Ageing prevention of VE is also controlled by design and by regular flushing during periodic testing. The monitoring programme completes the ageing management. The AMPs for VF, VE and UJ are considered to be effective.

There is no component that has been inspected, analysed, researched and assessed more than the Reactor Pressure Vessel. Licensee EPZ has participated in different international programmes and is still doing so. The most important ageing issue is the potential embrittlement of the material. During the LTO-programme again a complete ageing management review of all aspects was done, including again introduction in 2007 of test specimens in the available irradiation chambers. With the conservative assumption of 55 FPY produced electricity at 60 years of the life the calculated fluency is below the assumed value for the design in 1973. Taking into account the first specimen assessments after 7 years (2014) at 60% of the 55 FPY, the RT-limit values are well below the norm, confirming a large margin. In 2018 the specimens have reached 100% and again the effects will be assessed. Based on foreign experiences like Davis Besse, the hydrogen flakes in Doel/Tihange and the Carbon Segregation issue additional inspections/measurements were done, that showed that these issues are not impacting the RPV. Also recent a modernized measurement was done to detect underclad cracks. None were found.

The purpose of the concrete dome surrounding of the primary steel containment is to protect the primary steel sphere from external impacts due to earthquake, explosion pressure waves, private aircraft crashes or external flooding. The ageing management is based on a list of ageing mechanisms compiled by experts of the OEM, containing the latest insights. A concrete inspection programme is available for the foundation, inner and outer walls, roofs, floors as well as coatings. E.g. flaws in concrete are inspected visually and using a chipping hammer. If needed further analysis is carried out. A whole set of analyses is described. If needed repairs are carried out. For instance in several places the floor coatings were considered at the end of their life and were replaced.

9.3 Overall assessment and general conclusions - Commonalities at all RRs

Like with the NPP, for the research reactors in the past all kinds of maintenance activities could be called ageing management activities.

It can be concluded that although the HOR started later with the development of an AMP according to modern standards than the HFR, its AMP is in a more advanced state today. This is due to the smaller size, simpler reactor system and lower risk of the reactor to be considered. On the other hand the inclusion of LTO in the HFR programme is an additional step and needs more effort. Still the relatively short time of development of the AMP of the HOR could be considered as a good practice.

Both research reactors have carried out an AMR and described the AMP for cables and wires, obviously with a different approach. It can be concluded that in general the ageing of cables of the research reactors is under control.

Concealed piping only is applicable to the HFR.

9.4 Overall assessment and general conclusions - HFR-specific information

The HFR was stimulated by ANVS to develop ageing management programmes around 2005 through the first PSR. A licence requirement was introduced (requiring a roadmap), as well as the requirement for having an INSARR mission (which includes an AM module) every 5 years. The first ageing management developments were based on IAEA standards for RR and NPP. The INSARR missions in 2005, 2011 and 2016 all looked at AM and produced recommendations. After a number of ageing related issues 2008-2013, with large impact on the operation of the reactor and a stepwise increased ANVS inspection regime, the licensee started in 2013/2014 an Asset Integrity Programme and AMR according to the new IAEA SSG-10. Structurally more manpower was put in maintenance.

Based on the AMR, improvements have been carried out since 2015-2016 in the existing AM-system, but much still has to been done at the HFR to reach the ambitious goal to complete the integrated AMP and fully comply with IAEA SSG-10 and to be ready for an SCO-mission in 2019/2020. This work has started in 2017. The recruitment, in the beginning of 2017, of a new maintenance manager, especially with experience from the non-nuclear industry in the development of these kinds of programmes, is considered as a good practice. Today the new approach has not yet been implemented in the management system. That is part of the improvement process.

As part of the AMP for cables, in 2014, a cable depot was introduced at the HFR. This might be a candidate for a good practice for a research reactor. In general not much degradation was found, but several cables have to be included in the programme. The most ageing sensitive are the coax cables to the neutron flux measurements in the reactor pool. They will be replaced as a preventive measure. The licensee has found that the most safety relevant cables that are not yet under a degradation monitoring programme are the redundant underground power cables from the diesel generators to the HFR (there is only a detection to determine if the cable is in operation). These will be included in the programme.

In the HFR there are a number of concealed pipes that have a leak before break design and an associated detection mechanism. A number of pipes are partly inaccessible and therefore sometimes difficult to inspect. The ageing management programmes for these pipes are now under development. In a buried tritium transport pipe a leakage was discovered in 2012. This pipe had no leak detection, nor was it on the maintenance list. It has been replaced by an above ground pipe, surrounded by another pipe, with leak detection. Currently all other relevant underground pipes are part of a monitoring programme. The case of 2012 was one that was a wake-up call prompting for increased activity to improve ageing management.

9.5 Overall assessment and general conclusions - HOR-specific information

At the HOR the first steps to develop an AMP were made through the PSR. The AMP was developed in 2012-2015 based on IAEA SSG-10. Experience exchange was done with Belgian BR2 and South-African SAFARI. In 2016 ANVS approved the AMP as having been implemented. It has been documented in the management system. One important improvement that was introduced is the learning form internal and external experience. Many new periodic/preventive maintenance actions have been determined. The newly developed database is according to the licensee essential to keep track of upcoming periodic tests. In 2020 after about 20 years there will be a second INSARR mission, coupled to the PSR.

With regard to cabling, the situation at the HOR differs from that at the HFR. This is due to the fact that the HOR does not need electricity for the preservation of the three main safety functions. A corrective maintenance strategy is therefore applied. Many cables were replaced 35 years ago when the control room was moved to the outside of the reactor building. Newly specified cables taking account of the environmental circumstances were installed that still function very well. After the AMR several other cables have also been replaced. In 2009 moisture intrusion into the cable for the fission chamber was detected and remedied by a periodic action to flush the housing.

Appendix A. HFR Vessel information

A.1. Introduction

The HFR does not have a steel RPV. A description of the ageing management procedures for the HFR reactor vessel falls out of scope of the TPR. Nevertheless the reactor vessel is one the most safety relevant SSCs of the HFR, which is reflected in the extensive programmes that exist to monitor the condition and predict the ageing behaviour of the vessel. A description of ageing management at the HFR would not be complete without addressing these programmes.

A.2. Description of the HFR reactor vessel

The HFR reactor vessel, shown in the Figure below, is a complex set of box like structures welded together from plates, pipes and forgings. The vessel is made of aluminium alloy 5154-O and has been in placed in 1984, when the original reactor vessel was replaced. The reactor core box is situated just above the cylindrical part of the structure. The chimney structure is placed on top of the core box. The rack supports the poolside facility west (PSF-I).

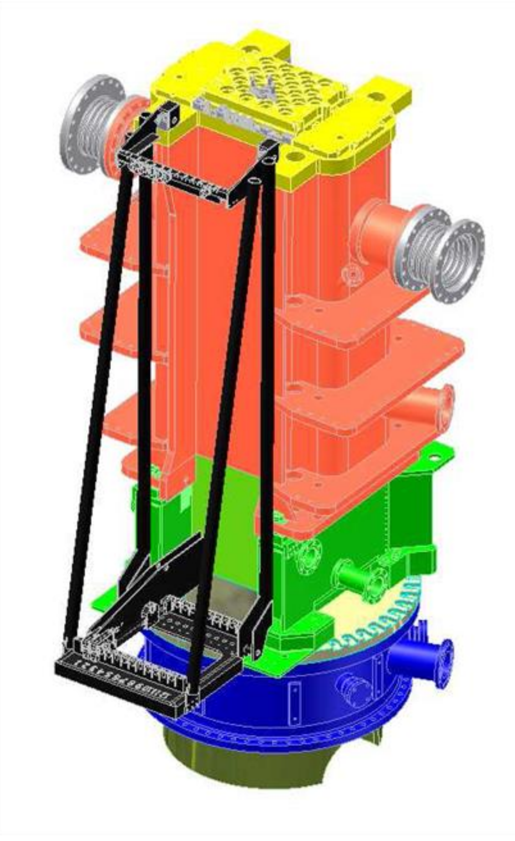

Figure 34 The present HFR vessel design and related structures

The main functions of the HFR reactor vessel are:

- to provide guidance of the cooling water through the core with the heat generating fuel elements;
- to provide the structure and structural integrity needed to maintain the fuel element configurations and to control rod movements under normal and abnormal conditions;

- to provide neutron beams to the instruments outside the pool.

A.3. Ageing mechanisms

The vessel is exposed to thermal and fast neutron flux and is in contact with water at an operating temperature of 40-66 ^{o}C. The following three mechanisms are potentially relevant damage mechanisms which can induce changes to the reactor vessel material properties during its operational life:

1. Thermal ageing
2. Corrosion
3. Neutron radiation damage

Thermal ageing is the phenomenon of the microstructural changes in the material due to new phase formations at elevated temperatures leading to properties degradation. For the non-hardened aluminium alloy 5154-O this is a highly unlikely mechanism of degradation as the operational temperature of HFR ranges from 313 to 338 K (40°C to 65°C). Thermal ageing of the specimens is prevented by the condition that the maximum specimen temperature in the surveillance rigs remains below 100 °C.

Corrosion is also an unlikely contribution to the degradation of 5154-O alloy. This material has been developed for good corrosion properties under restriction that the water quality is well controlled within the prescribed limits. If significant corrosion occurs it will be detected as part of regular in-service inspections.

Neutron damage can occur in the form of fast neutron displacement damage, gaseous transmutation damage caused by the formation He and H through (n, α) and (n, p) transmutation reactions and thermal neutron damage caused by transmutation of Al into Si through the following sequential reactions:

$^{27}Al(n, \gamma)^{28}Al$, $^{28}Al \rightarrow {}^{28}Si + \beta$

leading to an increase in Si content with increasing thermal neutron fluence. This is the dominant ageing mechanism in the vessel material leading to irradiation hardening and embrittlement.

To guarantee that the vessel can perform its safety functions at all times, the ageing of the vessel is monitored in two programmes. Firstly, the in-service inspection programme is aimed at performing non-destructive examinations of the entire vessel. Secondly, the surveillance programme focusses on the effects of neutron induced ageing of the core box wall.

A.4. Monitoring and inspection activities

In service inspection programme

The in-service inspection consists of non-destructive inspection of the vessel during reactor outages. The programme has started in 1984. The inspection programme consists of:

- Volumetric inspection (e.g. ultrasonic testing) of selected welds in the vessel. This part aims for defect detection in the vessel and especially the welds. Reportable indications are checked against the allowable defect size;

- Surface inspection (e.g. visual inspection, eddy current testing) of the vessel wall and welds for defect detection at and near to the vessel surface, particularly checking for surface breaking defects.

A dedicated inspection programme is developed based on the regulations for nuclear power plants in the ASME Boiler and Pressure Vessel codes (ASME XI), adapted for use in a research reactor such as the HFR. The inspection interval and qualification of staff, procedures and equipment are in line with internationally accepted codes and standards (ASME XI, ISO9712, INEQ).

The inspection programme has been approved by the nuclear regulatory body and the programme is executed under supervision of the accredited notified body (Lloyd's Register).

Surveillance programme

To address the ageing of the core box, the region exposed to neutron radiation, a dedicated surveillance programme has been set up when the replacement vessel was installed in 1984. Both in core and out of core surveillance specimens (full size CT specimens and tensile specimens) are placed in irradiation facilities aimed at ageing in a manner representative for the vessel wall. The surveillance specimens are machined from the same batch of aluminium as the core box itself.

The part exposed to a high neutron flux is relatively small: the core box is an approximately square box of about 0.9 x 0.7 m^2, with 50 mm wall thickness. The active fuel of the reactor core is 0.6 m high, and the neutron flux decreases rapidly above and below it. The locations that receive the highest thermal flux are located very near the middle of the core box sides; the core box side on the Westside (PSF-I) is subjected to the highest thermal flux and is referred to as the hot spot.

The vessel design report provides the allowable stress levels under normal and abnormal conditions. The design process for the reactor vessel, a main primary nuclear component of the HFR is well specified in the ASME III standard. In addition to the stress analyses the design report specifies allowable defect sizes in accordance with the ASME XI rules. Relevant for the HFR Surveillance Programme is that the design report specifies that the minimum allowable fracture toughness of the HFR reactor vessel is 6 MPa√m.

Approximately every two years since the installation of the vessel one or two surveillance specimens have been extracted from the irradiation facilities and tested. The tests are performed in accordance with international standards (ASTM E8M for tensile testing and ESIS P2-92 or ASTM E399 for fracture toughness testing depending on the fracture mechanics).

Although the aim is to ensure the surveillance specimens are subjected to a higher thermal neutron fluence than the vessel wall, it is very difficult to achieve this in a research reactor such as the HFR, where the vessel is placed directly around the core fuel and is surrounded by thermalizing water. As a result, the surveillance specimen with the highest fluence has seen slightly fewer neutrons than the hot spot in the vessel wall. To make the programme fully predictive the measurements on the surveillance specimens are complemented by an extensive scientific endeavour to extrapolate the measured mechanical properties. This activity consists of literature studies and microstructural examinations and investigations. The findings of these activities are shared with the international community via coordinated research projects of the IAEA.

As of 2017, the fracture toughness of the hot spot region of the core box wall is determined to be 20 – 25 MPa√m and from the scientific investigation it is not expected to drop fast in the coming years. Therefor it is believed that the vessel wall will not reach it design limit of 6 MPa√m within the remaining lifetime of the HFR.

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