Summary report Ageing Management Review

Under the contract of N.V. EPZ

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Summary

The Borssele Nuclear Power Plant (Kerncentrale Borssele, KCB) plans to extend its operating life to 60 years, until 2034. To demonstrate that safety and safety relevant Systems, Structures and Components (SSCs) continue to perform their intended functions during Long Term Operation (LTO) KCB started the project “LTO Bewijsvoering” (LTO justification).

As part of this project an Ageing Management Review (AMR) was performed by AREVA in cooperation with KCB. In the AMR the conditions of all passive in-scope SCs were assessed and a technical evaluation to demonstrate that their intended functions will be managed consistent with the current licensing basis during LTO was performed. The AMR for structural components concluded that all structural SCs are adequately managed with the existing ageing management activities. However the mechanical, as well as the electrical AMR resulted in several recommendations. The current report summarizes these recommendations and describes the way KCB intends to implement them. With the fulfillment of these recommendations the effects of ageing on in-scope SCs will be adequately managed, so that their intended functions will be maintained consistent with the KCB licensing basis for Long-Term Operation (LTO). In addition, with the implementation of these recommendations, KCB plant practices and policies are in alignment with accepted nuclear industry practices for LTO.

List of abbreviations

AMR	Ageing Management Review
BAC	Boric Accid Corrosion
CRDM	Control Rod Drive Mechanism
CUF	Cumulative Usage Factor
ECP	Electrochemical Potential
FAC	Flow Accelerated Corrosion
GAO	Gebruiks Afhankelijk Onderhoud (time-based maintenance)
H-AVT	High All Volatile Treatment
HISCC	Hydrogen Induced Stress Corrosion Cracking
HP	High Pressure
HVAC	Heating Ventilation and Air-Conditioning
IAEA	International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
IGC	Intergranular Corrosion
IGSCC	Intergranular Stress Corrosion Cracking
ISH	Instandhouding
ISI	In-Service Inspections
LP	Low Pressure
LTO	Long Term Operation
LWR	Light Water Reactor
KCB	Kerncentrale Borssele (NPP Borssele)
MCL	Main Coolant Line
MCP	Main Coolant Pump
MCPB	Main Coolant Pressure Boundary
MIC	Microbiologically Influenced Corrosion
NPP	Nuclear Power Plant
ODSCC	Outer Diameter Stress Corrosion Cracking
OEM	Original Equipment Manufacturer
POD	Probability Of Detection
PRV	Pressure Relief Vessel

PRZ	Pressurizer
PSA	Probabilistic Safety Assessment
PT	Penetrant Testing
PWSCC	Primary Water Stress Corrosion Cracking
PWR	Pressurized Water Reactor
RPV	Reactor Pressure Vessel
SC	Structure and Component
SG	Steam Generator
SICC	Strain Induced Corrosion Cracking
SL	Surge Line
SSC	System, Structure and Component
TGSCC	Transgranular Stress Corrosion Cracking
TÜV	Technischer Überwachungs-Verein (Technical Inspection Association)
UT	Ultrasonic Testing
VGB	Verband der Großkessel Besitzer
VT	Visual Testing
WANO	World Association of Nuclear Operators

1 Introduction

1.1 Framework

The Borssele Nuclear Power Plant (Kerncentrale Borssele, KCB) plans to extend its operating life to 60 years, until 2034. To demonstrate that safety and safety relevant Systems, Structures and Components (SSCs) continue to perform their intended functions during Long Term Operation (LTO) KCB started the project “LTO Bewijsvoering” (LTO justification). This project is described in the conceptual document [4]. The outline of this project is based on IAEA Safety Report 57 “Safe Long Term Operation of Nuclear Power Plants”[1]. A schematic overview of the LTO project, following this guideline, is shown in Figure 1-1.

The identification of SSCs that are within the scope of LTO assessment was performed during the scoping [2] and subsequent screening [3] processes. During the scoping process (4.1 in Figure 1-1) the safety functions of all SSCs were analyzed in detail, and subsequently the SSCs were categorized in three different “safety categories”, which are in accordance with IAEA draft Safety Guide DS367[5] combined with AREVA experience and engineering judgement:

- Safety category S1 contains components of the reactor coolant system whose postulated catastrophic failure is not enveloped by accident analyses.
- Safety category S2 contains:
  - High-energy components inside the containment whose postulated failure may lead to cross-redundancy consequential damage and is not enveloped by accident analyses, where such components are not already assigned to safety category S1.
  - High-energy components inside the reactor building, whose failure initiate a design-basis accident with immediate adverse impact on heat removal from the reactor core.
  - Components required for the control of design-basis accidents (safety functions), for whose function alternative measures are not available promptly or in an adequate time frame.
  - Components of auxiliary/supply functions whose failure leads, causally and in the short term, to loss of safety functions required for accident control.
  - Supports as well as supporting structures for S1 components.
- Safety category S3 contains SSCs, whose failure may impact upon the safety functions specified in categories S1 and S2.

The scoping results are limited to a system, subsystem or building level [2].

In the subsequent screening process (4.2 in Figure 1-1) the active and passive Structures and Components (SCs), that support the SSCs classified as safety category S1-S3 in the screening process, and their intended safety function(s) were identified using a commodity group approach [3]. Active SCs are defined as structures, components or subcomponents that perform or support an intended safety function in support of the structures or component /commodity group function(s) with moving parts or with an intended change in configuration or state. Passive SCs are defined as structures, components or subcomponents that perform or support an intended safety function in support of the structures or component /commodity group function(s) without moving parts or without an intended change in configuration or state e.g., piping or vessels [3].

In the Ageing Management Review (AMR) all passive SCs identified as in-scope for LTO were evaluated (green box in Figure 1-1). The AMR involved a detailed technical evaluation of safety and safety-related, passive components (e.g., piping) and subcomponents (e.g., pump casing) to demonstrate that the effects of ageing will be adequately managed, (i.e., the intended function(s) will remain consistent with the NPP licensing basis during LTO).

The AMR is divided in 4 different parts (see section 1.3):

- Mechanical A;
- Mechanical B;
- Electrical and I&C;
- Structural.

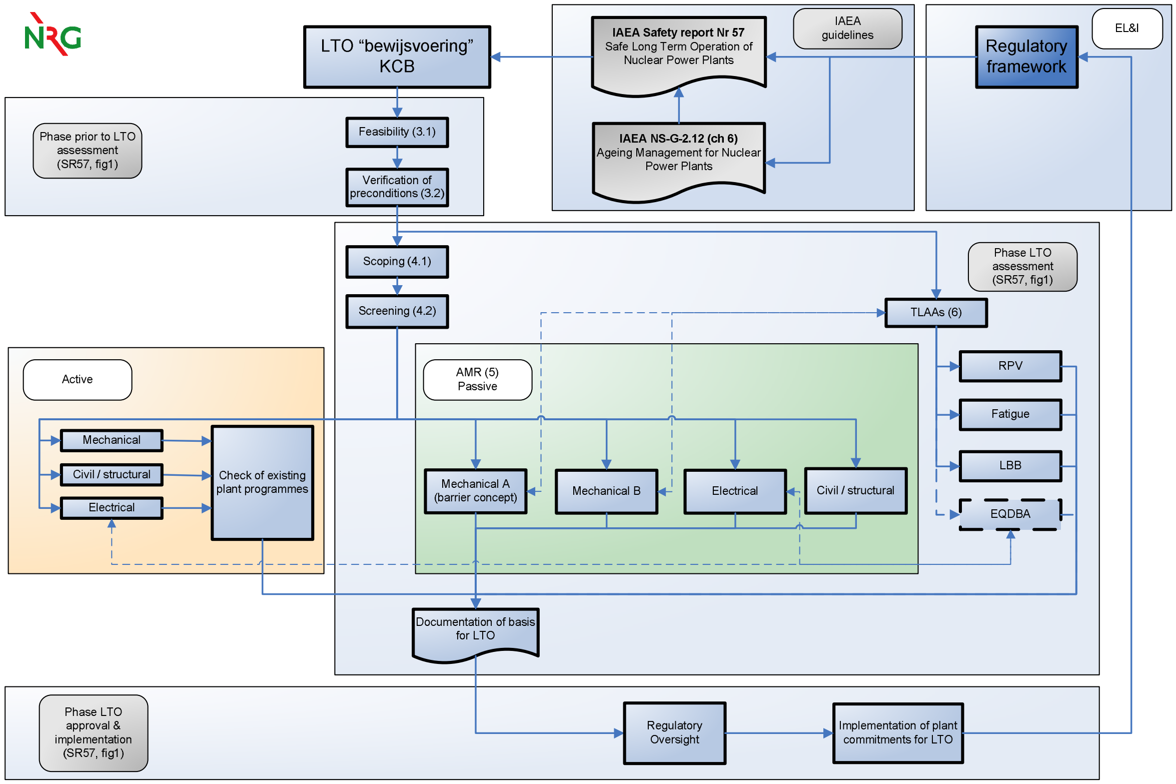

Figure 1-1 Schematic overview of LTO Bewijsvoering [4]

1.2 AMR methodology

In the AMR the conditions of all passive in-scope SCs were assessed and a technical evaluation to demonstrate that their physical status (i.e. their intended functions) will be managed consistent with the current licensing basis during LTO was performed. The outline of the AMR is based on IAEA Safety Report 57 [1]. This guideline describes the following three AMR steps (Figure 1-2) [1]:

- Step 1: Identification of ageing mechanisms that require management.
- Step 2: Evaluation of relevant ageing mechanisms; are they adequately managed.
- Step 3: Modification of existing ageing management activities/Introduction of new ageing management activities.

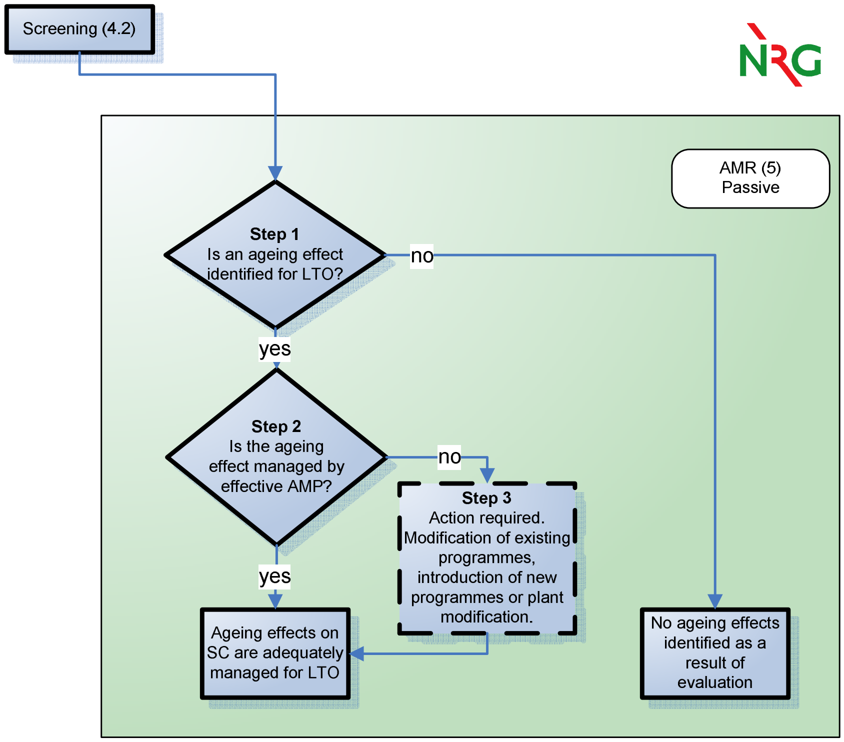

Figure 1-2 Overview of Ageing Management Review (AMR) process according to SR57 [1] [4]

A general description of each AMR step is provided below. A detailed description of the AMR methodology is provided in the Ageing Management Review - Methodology Report [6].

1.2.1 Step 1: Identification of ageing mechanisms that require management

In this step the ageing mechanisms that require management for each in-scope SC were identified and assessed with respect to their likelihood during a 60 year service period. This review of relevant ageing mechanisms is based on the environmental and operating conditions for each in-scope component and subcomponent, component material information, as well as site-specific and industry operating experience and relevant research experience. Component materials frequently used at KCB include steel (e.g., carbon steel, low-alloy steel, chrome-molybdenum steel), cast iron, stainless steel, nickel-based alloys, copper alloys, aluminium and non metallic's. The environmental and operating conditions include chemistry conditions, operating temperatures, operating pressures, flow rates, medium type and other relevant environmental conditions for all modes of plant operation.

To facilitate step 1 a catalog of ageing mechanisms was prepared for the disciplines mechanical [7], electrical and I&C [8] and structural [9] (see Figure 1-3). These catalogs address ageing mechanisms that are identified to be potentially relevant for KCB and provide a general description of relevant mechanisms, as well as basic information regarding, conditions of concern, components/areas of concern, examination and detection, prevention, mitigation and remedies and examples.

1.2.2 Step 2: Evaluation of relevant ageing mechanisms; are they adequately managed

Once the review of relevant ageing mechanisms was completed, the necessity for relevant ageing management activities was identified. Effective ageing management can be accomplished by coordinating existing measures or activities, including maintenance, In-Service Inspection (ISI) and surveillance, as well as operations, technical support programs (including analysis of any ageing mechanisms) and external programs, such as research and development. In this step existing plant activities were identified, reviewed and evaluated to determine if the existing activities are adequate without modification or whether existing activities should be augmented for LTO.

1.2.3 Step 3: Modification of existing ageing management activities/Introduction of new ageing management activities

The evaluation of relevant ageing mechanisms may show that the existing KCB activities are adequate to manage ageing. However, if an identified measure or activity is not adequate or does not exist at KCB recommendations were made regarding the specific areas in which KCB plant practices and policies should be augmented to substantiate Long Term Operation.

1.3 AMR breakdown

As outlined in the AMR methodology report [5] a differentiation in the groups:

- Mechanical;
- Electrical and I&C;
- Structural.

was made for the AMR.

The mechanical passive in-scope SCs were further differentiated between mechanical A and mechanical B SCs. The mechanical A SCs were defined as forming a part of the fission product barrier. According to the defence in depth concept there are four physical barriers that prevent the release of radioactivity to the environment. The first and second physical barrier are covered by the fuel assemblies. As the fuel assemblies are replaced on a regular basis they are not subject to an AMR. So the mechanical A SCs constitute the third (main coolant pressure boundary) and fourth (containment system) physical barrier.

The mechanical A AMR considered the passive subcomponents within the following SCs:

- Reactor Pressure Vessel (RPV);
- Pressurizer (PZR);
- Steam Generators (SGs);
- Main Coolant Pumps (MCPs);
- Control Rod Drive Mechanism (CRDM) pressure housings;
- Main Coolant Lines (MCLs) and pressurizer surge line;
- Steel containment.

The mechanical B SCs consist of the remaining in-scope mechanical systems. The in-scope mechanical B SCs were handled in four system groups:

- Nuclear safety systems;
- Safety-related auxiliary systems;
- Secondary systems;
- Heating, Ventilation and Air-Conditioning (HVAC) systems.

and in addition four further groups:

- RPV internals;
- Primary component supports;
- Remaining in-scope supports and hangers;
- Mechanical fasteners.

For each of above listed (sub)groups an AMR report was prepared (see Figure 1-3) according to the methodology outlined in section 1.2.

As stated in section 1.2.1 catalog of ageing mechanisms were prepared to facilitate “Step 1: Identification of ageing mechanisms that require management” for the disciplines mechanical [7], electrical and I&C [8] and structural [9].

As shown in Figure 1-3 the results of the AMR are gathered and summarized in the current report.

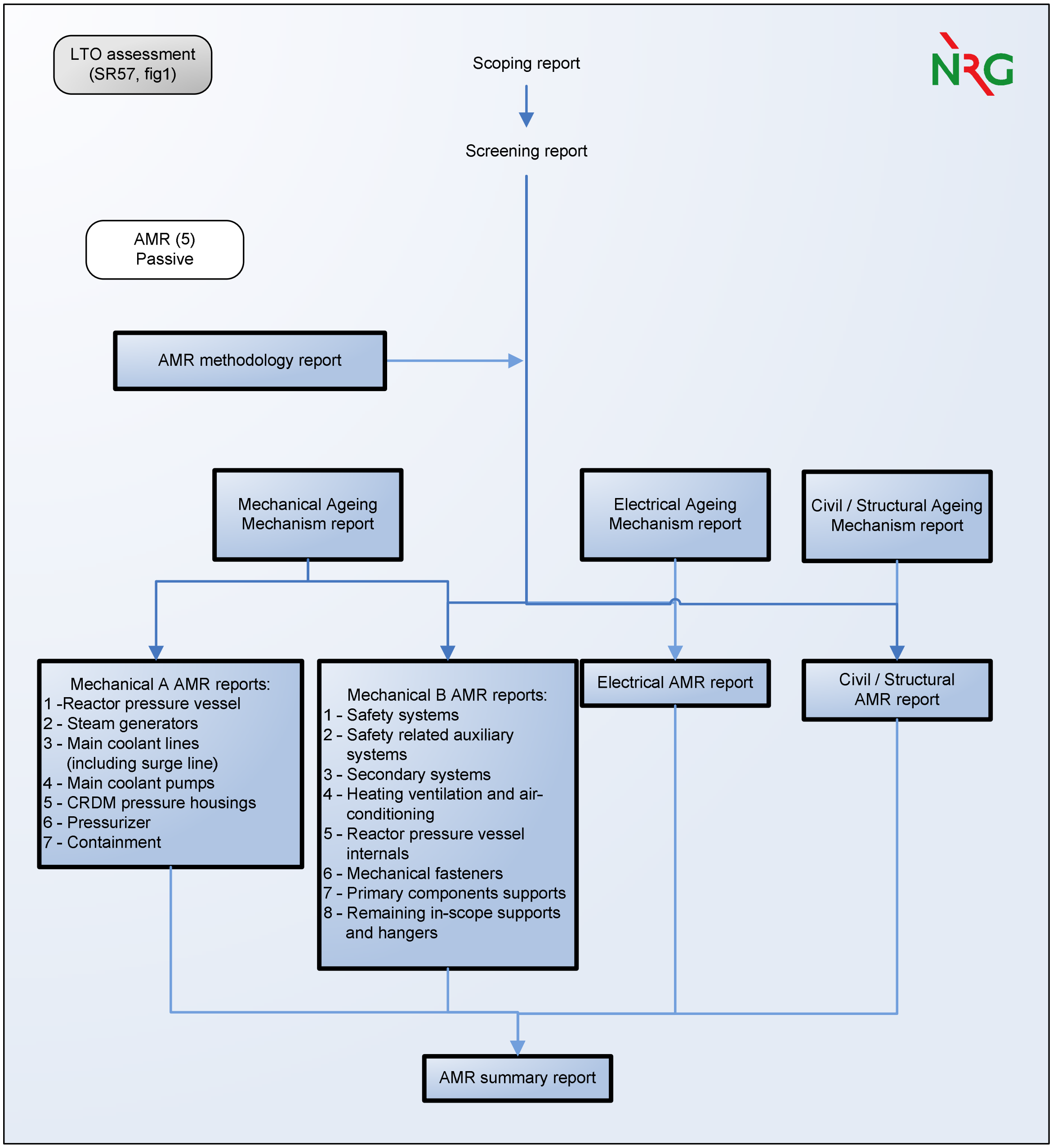

Figure 1-3 Breakdown of the AMR [4]

1.4 Current report

The purpose of the current report is to summarize the results of the mechanical, electrical and I&C and AMR for structural components and to describe how the AMR recommendations will be implemented at KCB (Figure 1-3).

Chapter 2 of this report provides an explicit statement on the current physical status of the plant SCs. In the following chapters the results of the technical evaluation to demonstrate that the effects of ageing will be adequately managed (i.e. the AMR) are described. The AMR has been performed by AREVA in cooperation with KCB. Chapter 3 describes the results of the mechanical AMR, chapter 4 the results of the electrical and I&C AMR and chapter 5 the results of the AMR on structural components. In case the AMR identified that an existing ageing management activity was not adequate or does not exist at KCB, these chapters describe the recommendations made by AREVA regarding the specific areas in which KCB plant practices and policies should be augmented. Chapter 6 describes how KCB intends to implement these AMR recommendations.

2 Current physical status

Ageing is predominantly caused by intrinsic factors (e.g., operating temperature, pressure, neutron radiation) which cannot be eliminated because they are an integral part of the processes that occur within a Nuclear Power Plant (NPP). As intrinsic factors are generally well known, they are considered during the design phase and/or are taken into account through activities/measures during operation. To ensure that the current physical status of the plant, despite ageing, is within the predicted scope of the design, and therefore adequate for continued plant operation, Safety Report 57 [1] requires an engineering assessment of the current physical status of SCs.

An adequate engineering assessment requires knowledge of [1]:

1. The design, including applicable codes and regulatory requirements, the design basis and design documents, including the safety analysis;
2. The fabrication, including material properties and specified service conditions;
3. The operation and maintenance history, including commissioning, operational transients and events, generic operating experience such as power uprating, modification and replacement, surveillance and trend curves;
4. The results of inspections;
5. The environment inside and outside the SCs;
6. Results of research.

These topics have all been taken into account in the assessment of KCBs ageing management activities as described in the AMR reports [6]. Based on this assessment, the AMR reports do provide statements on the adequacy of the management of all relevant ageing mechanisms for passive, safety relevant SCs during LTO. However an explicit statement on the current physical status of these SCs, based solely on the information that has been collected for the performance of the AMR, has not been provided.

The consideration of ageing during design and fabrication of the plant (items 1 - 2) is presented in section 2.1 of this report. It will be argued that the processes and qualified procedures followed at the time of erection ensured that the physical status of the plant was well described and that deficiencies have been documented to ensure that ageing can be adequately managed during operation.

Ageing management during plant operation (items 3 - 6) is explained in section 2.2. The current KCB plant programs and their interrelations are presented. The continuous improvement of these programs through evaluation of inspection results and through knowledge of internal and external operating experience and research is also addressed.

Section 2.3 highlights relevant modifications, operation and maintenance history, including inspection results, and external operating experience and research of SCs. Based on the generic statement on the documented status of the plant after erection (section 2.1), the consideration of relevant modifications, operating and maintenance history, external operating experience and research on specific SCs, combined with the argument that inspection results from current plant programs do not show any unacceptable deviations and the extensive information in the AMR reports, a statement on the current physical status is provided.

2.1 Consideration of ageing during the design phase

Ageing was taken into account throughout the KCB design phase. KCBs structures and components needed to meet general construction and design requirements in accordance with Dutch requirements (Dienst voor het stoomwezen) [28], requirements of ASME section III of the boiler and pressure vessel code 1968 ("nuclear vessels") [29], and/or other requirements for SCs such as Siemens/KWU specific requirements, which are described in several specification documents (RE-Ls). Plant specific welding processes and qualifications were described in the associated Siemens/KWU "Arbeitsvorschriften (AVS)" and also referred to in the RE-Ls.

The relevant codes and requirements for the different stages of manufacturing and construction of each in-scope SC are summarized within the corresponding AMR report. Relevant requirements include:

- Calculation and construction requirements;
- Material requirements;
- Material testing requirements;
- Welding requirements and qualifications;
- Inspection and non-destructive examination requirements;
- Etcetera.

Inspection and acceptance of the manufacturing processes and products were performed by the German acceptance authority TÜV and/or the Dutch inspectorate "Dienst voor het stoomwezen". Multiple examinations were performed during manufacturing e.g., volumetric testing and chemical analysis. To confirm the physical status of (semi)finished products appropriate non-destructive testing was performed.

In the end, the Dutch inspectorate “Dienst voor het stoomwezen” and/or the TÜV performed the final acceptance according to the specifications and requirements of the “Dienst voor het Stoomwezen”

The processes and qualified procedures followed during the manufacturing and final acceptance of the plant ensured that the physical status of the plant at the time of erection was well described in, e.g., semi-finished product, welding and heat treatment documentation, and according to the “state of the art”. Any identified flaws were well documented to ensure that their acceptability was justified and that ageing could be adequately managed during operation.

2.2 Management of ageing during operation

Effective ageing management during operation is accomplished at KCB by coordinating existing plant asset management programmes (instandhoudingprogramma’s), including the preventive maintenance, in-service inspection and surveillance programs, as well as operations, and the evaluation of internal and external experiences. A general overview of KCBs asset management programs is presented in Figure 2-1.

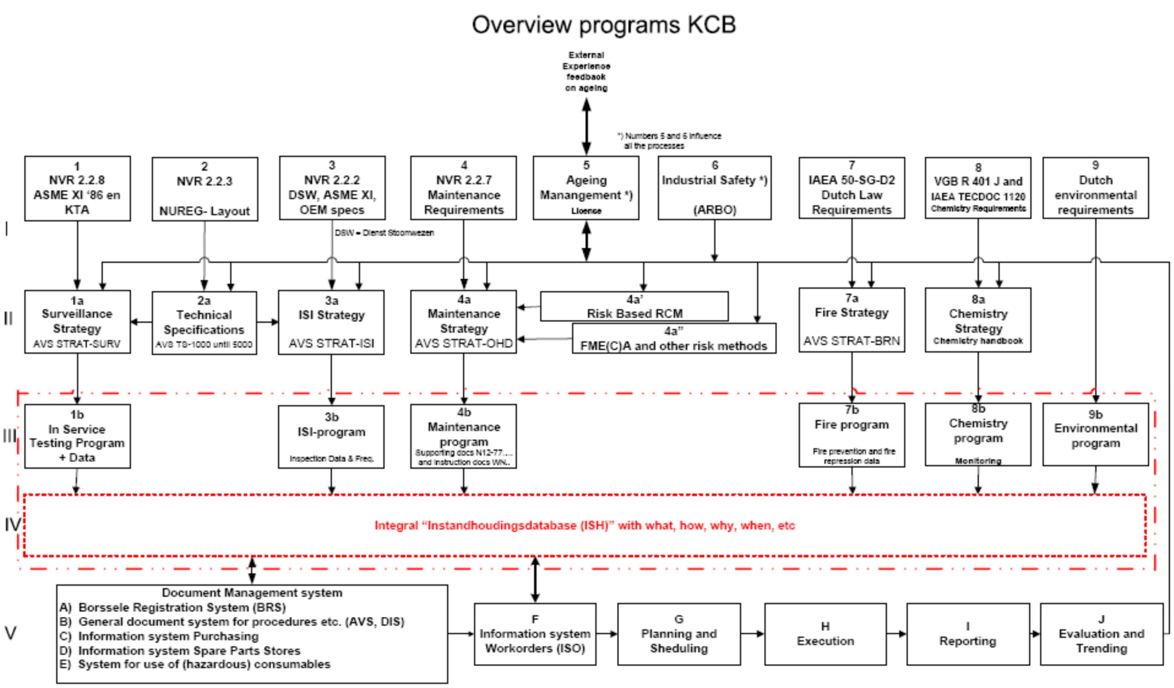

Figure 2-1 General overview of KCBs existing asset management programs

2.2.1 Preventive maintenance program

In accordance with KCBs maintenance strategy, safety relevant SSCs are to be maintained using preventive maintenance [30]. KCBs preventive maintenance program (item 4b in Figure 2-1) includes actions that detect, preclude, or mitigate the degradation of functional SSCs to sustain or extend its useful life by controlling degradation and failures to an acceptable level. It includes periodic replacement, time-based maintenance and condition-based maintenance. The results of surveillance tests, as well as the evaluation of internal and external operating experiences are used as inputs to define the scope of preventive maintenance activities [31].

2.2.2 ISI program

The ISI program (item 3b in Figure 2-1) monitors the ageing effects of selected safety class 1, 2 and 3 SSCs [31]. ISI is performed on components to provide continuing assurance that they will safely operate throughout the lifetime of the plant, contributing to plant availability and minimizing the probability that an event will occur. KCB uses qualified and code-approved inspection methods and techniques, based on critical flaws and a Probability of Detection (POD) with adequate margin. ISI activities are performed at KCB on an ongoing basis, throughout the service life of the plant. They allow the comparison of pre-service and baseline information to assess changes or degradation in equipment condition which have occurred with time. In addition ISI activities provide a sound basis for the inspection program, suitable for LTO.

KCBs current ISI program for nuclear components considers the requirements of Stoomwet and ASME code section XI [33], taking specific design elements of the plant into account. The results of former studies, inspection results and internal and external operating experience have been used to modify the scope of inspections, as agreed between the Dutch regulator and KCB.

KCBs ISI program uses three types of examination [34]:

- Volumetric examinations; ultrasonic testing (e.g., pulse echo and time of flight diffraction technique), Eddy current examination or radiographic/X-ray testing to detect volumetric defects within materials.

- Visual examinations:
  - VT-1: Visual testing is used to determine the surface condition for defects or degradation effects, such as cracking, erosion, abrasion, corrosion, wear. In some cases, the component can be disassembled for examination.[1]
  - VT-2: Visual testing is used to identify leakage during hydrostatic pressure testing and prior to start-up of the plant after refueling outages. The VT-2 examinations prior to start-up are performed at full operating pressure and temperatures during a plant walkdown.
  - VT-3: Visual testing is used to determine the general mechanical and structural condition of a component and its supports, or it is used to detect discontinuities and imperfections such as loss of integrity at bolted or welded connections, or it is used to observe conditions that could affect operability or functional adequacy of constant load and spring-type components and supports. The plant walkdown prior to startup after each refueling outage considers VT-3 as well.
- Surface examinations; ultrasonic testing (creep wave technique), magnetic particle examination, liquid penetrant examination, or eddy current examinations to indicate the presence of surface discontinuities and flaws (e.g., cracks).

2.2.3 Surveillance program

The objective of the KCB surveillance program is to ensure the functional availability of systems and components that perform safety functions, within operational limits and conditions, to promptly detect SSC deterioration, as well as any trend that could lead to an unsafe condition. The surveillance program includes the performance of periodic inspections and testing to establish and direct monitoring, inspection and maintenance of in-scope SSCs, as required by regulations or requirements [35] [36].

Surveillance generally involves the following activities :

- Monitoring of operating parameters, such as pressure, temperature, flow rates, and level measurements and system status. This includes:
  - Water chemistry program [37] (item 8b in Figure 2-1) ;
  - Vibration monitoring;
  - Leakage monitoring;
  - Loose parts monitoring;

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[1] In addition KCB performs visual inspections using a robotic submarine. These inspections are regarded as equivalent to a VT-1 examination.

- Fatigue monitoring system (FAMOS)

- Checking and calibration of instrumentation.
- Testing and inspection of SSCs, including In-Service Testing (IST) (item 1b in Figure 2-1) and plant walkdowns.

  Plant walkdowns

  KCB operators perform walkdowns along the control panels, across the plant and through the buildings at the beginning of each shift with the objective of early detection and resolution of problems in systems and buildings across the entire NPP. Additionally KCB department managers perform regular plant walkdowns and bi-annual multi-disciplinary (WANO) plant walkdowns and the KCB maintenance department performs plant walkdowns in order to evaluate the physical condition of plant, e.g., to plan preventive maintenance or to assess the necessity for future improvements.

2.2.4 Operation

In order to minimize the rate of ageing degradation of plant SCs KCB operates according to prescribed procedures and technical specifications [41] (item 2a in Figure 2-1). These procedures and technical specifications include measures to restrict e.g., too high temperatures and/or temperature transients.

2.2.5 Evaluation of internal and external experiences

KCB evaluates the inspection results recorded during maintenance, ISI and surveillance activities, as well internal and external experiences (including WANO/VGB operating experience feedback, containing general and specific Siemens/KWU design or operating topics). Ageing related topics are separately evaluated. Depending on the evaluation results it is determined whether measures are needed (see Figure 2-2).

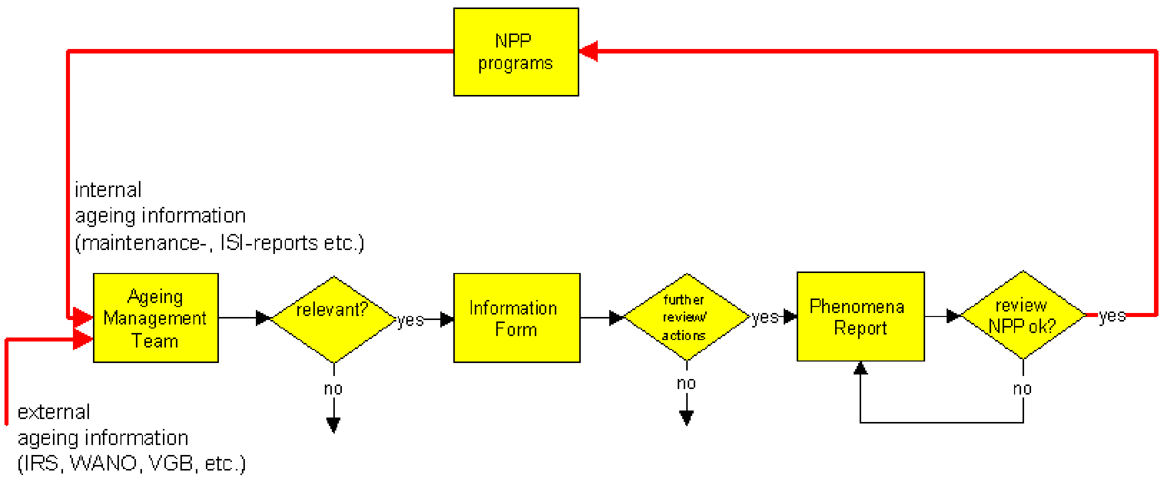

Figure 2-2 Evaluation cycle of ageing related internal and external operating experience

Besides these activities to continuously improve the organization and its processes, KCB conducts and reports a periodic safety review every 2 and 10 years. The 2-yearly safety review considers the safety performance against existing codes and standards. The objective of this review is to evaluate the technical, operational, human and administrative facilities with regard to nuclear safety and radiation protection, and to take appropriate action if consistency with the design basis cannot fully be demonstrated. The last 2-yearly safety review has been performed in 2011. This review showed no shortcomings in the provisions for nuclear safety and radiation protection. Some recommendations for improvement and suggestions for optimization of safety measures were presented [28].

The 10-yearly safety review evaluates developments in technological and regulatory insights to ensure that plant remains in compliance with the "state of the art". The first 10-yearly safety review resulted in the project "Modificaties" of 1997. The second 10-yearly safety review resulted in the project "Mod2Go". Both projects implemented safety measures, selected on the basis of their impact on the total core damage frequency or on the individual risk as calculated with the plant-specific living Probabilistic Safety Assessment (PSA) model. The evolution of the total core damage frequency and individual risk as a result of the safety improvements during the projects "Modificaties" and "Mod2Go" is shown in Table 2-1. The performance of a periodic safety review does not form a part of the LTO Bewijsvoering project (see Figure 1-1). The third 10-yearly safety review is currently underway.

Table 2-1 Evolution of the total core damage frequency and individual risk [39] [40]

Year	Total core damage frequency	Individual risk
1995	5,6 10^{-5}/jaar	8 10^{-7}
1997 (after project "Modificaties")	2,83 10^{-6}/jaar	2,6 10^{-8}
2009 (after project "Mod2Go")	2,12 10^{-6}/jaar	1,2 10^{-8}

2.3 Current physical status of SCs

The highlights of SCs relevant modifications, operating and maintenance history and external operating experience and research are summarized in this section. Based on this information, combined with the argument that inspection results from current plant asset management programs do not show any unacceptable deviations and the generic statement on the documented status of the plant after erection, a statement on the current physical status may generally be provided.

2.3.1 Current physical status of the RPV

In chapter 5 of the RPV and primary supports AMR [11] [10] relevant information to assess the current physical status of the RPV and its support is provided. The highlights of the RPV relevant modifications, operation and maintenance history, including results of ISI, and the external operating experience and research are presented below:

Some modifications have been performed on components within the scope of the RPV AMR in the past. The design of the intermediate extraction line has been improved. The new design prevents condensation in this line of any medium that leaked from between the two O-rings of the RPV head. This modification has no safety relevant effect on the RPV. Other modifications are the installation of level measurement assemblies and the use of newly designed high torque fasteners with improved seal quality for instrumentation and level measurement fasteners.

Transients, which could lead to changes in mechanical and thermal loading of materials, occur during plant operation. During the design phase of the plant the number of expected transients has been taken into account. During the service life of the plant KCB regularly evaluates whether the actual operating experience is still in line with the original expectations. The actual number of transients of importance for the RPV is well below the anticipated number of transients assumed during the design phase. With regard to irradiation embrittlement, it is important to note that KCB changed its core loading strategy to a partial low-leakage core strategy after 10 cycles of operation. After 19 cycles KCB completely implemented the low-leakage core strategy. The implementation of the low-leakage core strategy limits the neutron and gamma radiation levels on the RPV and therefore limits changes in RPV material properties due to neutron embrittlement.

The RPV (sub) components and its support are regularly inspected. The results of these inspections, predominantly recorded during ISI activities, show no unanticipated degradation and no inacceptable indications. Besides the inspections listed in the preventive maintenance, surveillance and/or ISI-program KCB performed several additional inspections. These included for example:

- Ultrasonic inspections to exclude crack growth of the underclad cracks in the RPV.
- Inspection of Bretschneider-verschlusse valves and flanges to identify corrosion in response to external operating experience.
- One-time inspection of the RPV system to verify the position of the RPV and to check that no rotation of the RPV had occurred in comparison with the original measurements in 1972 and verification measurements in 1989 [12].

Based on the above discussed measures, in combination with the results of in-service inspections, properly documented in inspection reports and annual ISI reports, and the extensive information in the RPV and primary supports AMR [11] [10], it can be concluded that the current physical status of the RPV (sub) components and its support is within the predicted scope of the design.

2.3.2 Current physical status of the pressurizer

In chapter 5 of the PZR and primary supports AMR [13] [10] relevant information to assess the current physical status of the PZR-system is provided. The highlights of PZR-system relevant modifications, operation and maintenance history, including results of ISI, and the external operating experience and research are presented below:

The PZR-system has been substantially modified from the original design configuration. These modifications included, the replacement of 3 safety valves and 3 relief valves with 3 SEBIM compact tandem safety valves that were mounted to a new dome. The new dome was fitted to the top flange of the pressurizer relief vessel in order to match the revised steam relief piping configuration. The 6 nozzles in the upper head were converted to perform the spray function by outfitting them with spray lances and thermal sleeves. The heating elements and heating element covers were replaced with new ones due to failed heating elements during the first 10 years of operation. The new heating elements have a double-wall design and a lower electrical load. They are therefore more reliable than the previous ones.

Transients, which could lead to changes in mechanical and thermal loading of materials, occur during plant operation. During the design phase of the plant the number of expected transients has been taken into account in e.g., fatigue analyses. During the service life of the plant several design calculations needed to be revised due to several modifications. In order to assure that the original and revised design assumptions are still valid, KCB regularly evaluates the actual number of transients to assure that these are below the anticipated number of transients. The actual number of transients important to the PZR is still well below the anticipated number of transients assumed during the design phase.

The PZR (sub)components are regularly inspected. The inspection results, predominantly recorded during ISI activities, are evaluated in order to demonstrate the (sub) component is adequate for continued operation. No unanticipated degradation and no inacceptable indications have been found. Besides the regular inspections listed in the preventive maintenance, surveillance and/or ISI-programs, KCB performed several additional inspections; for example an additional inspection to determine the cause of sealing ball leakage of the SEBIM valves one year after replacement.

Based on the above discussed measures, in combination with the results of in-service inspections, properly documented in inspection reports and annual ISI reports, and the extensive information in the PZR and primary supports AMR [13] [10], it can be concluded that the current physical status of the PZR (sub) components and their supports is within the predicted scope of the design.

2.3.3 Current physical status of the steam generators

In chapter 5 of the SG and primary supports AMR [14] [10] relevant information to assess the current physical status of the SGs and their support is provided. The highlights of the SG relevant modifications, operation and maintenance history, including results of ISI, and the external operating experience and research are presented below:

Several modifications on the SGs have been performed in the past. KCB changed for example the design of its SG tube supports after leakage of the U-tubes occurred. Inspections showed that the leakage was caused by tube support failure and subsequent fretting damage of the U-tubes. Nowadays the tube supports incorporate special comb-shaped constructions in the U-bend region, restoring the original tube support and minimizing flow induced vibrations and fretting. Due to leakage indications, the secondary side manhole covers were replaced with thicker versions of the same material. The original reduced shank bolts were replaced by longer ones. Another important modification is the transition from phosphate treatment to High All Volatile Treatment (H-AVT) for the secondary side water chemistry. The use of H-

AVT minimized the formation of crud and reduced the progress of corrosion on the SGs secondary side significantly. Only a limited number of SG tubes were plugged during the service life of the plant after ISI inspections showed wall thinning of these tubes due to wastage corrosion from the early years with phosphate treatment or fretting damage. Chemical cleaning of the secondary side of the SGs was performed in 2004, reducing the amount of crud on the tube sheet and supports significantly.

Several design calculations needed to be revised during the service life of the plant due the modifications. In order to assure that the original and revised design assumptions are still valid KCB regularly evaluates the actual number of transients to assure that these are in line with the anticipated number of transients. The actual number of transients important to the SGs is still well below the anticipated number of transients assumed during the design phase.

The SG (sub)components are regularly inspected. The inspection results are evaluated in order to demonstrate that status of the (sub)component is adequate for continued operation. These inspections resulted in several modifications (see above). Besides these regular inspections KCB performed several additional inspections. These included for example additional inspections during several years on the secondary side of the steam generators in the area of the modified U-bend region ("comb-shaped construction") to confirm the elimination of fretting damage.

Based on the above discussed measures, in combination with the results of in-service inspections, properly documented in inspection reports and annual ISI reports, and the extensive information in the SG and primary supports AMR [14] [10], it can be concluded that the current physical status of the SG (sub) components, including their supports is within the predicted scope of the design.

2.3.4 Current physical status of the main coolant pump

In chapter 5 of the MCP and primary supports AMR [15] [10] relevant information to assess the current physical status of the MCPs and their support is provided. The highlights of the MCP relevant modifications, operation and maintenance history, including results of ISI, and the external operating experience and research are presented below:

Several modifications have been performed at the MCPs since the construction and commissioning phase. In 1984 the hydrodynamic mechanical seals, which have a rotating hard metal ring and a static carbon ring and are assembled inside the two-stage seal insert in the seal housing, were modified. A wedge ring was replaced by an O-ring and therefore some parts of the seal ring support needed replacement. The rotating hard metal rings still have the same composition, but the manufacturing was improved, resulting in an increased lifetime of these slide rings. Another modification was the replacement of Flexitallic® IT 400 (containing asbestos) casing gaskets by graphite-containing gaskets in the eighties. This modification minimized possible chloride sources and thereby minimized the susceptibility to corrosion of applicable SCs.

Original design calculations, as well as a complete stress analysis of the casing, cover, cover studs and nuts, were performed by Sulzer for design basis transients to demonstrate that the subcomponents can withstand specified loads. In order to assure that the assumptions in the original design calculations are still valid KCB regularly evaluates the actual number of transients to assure that these are in line with the anticipated number of transients. The actual number of transients important to the MCPs is still well below the anticipated number of transients assumed during the design phase.

The MCP (sub)components are periodically inspected using e.g., Ultrasonic Testing (UT), Liquid Penetrant Testing (PT) and Visual Testing (VT). No unacceptable indications were found for in-scope MCP (sub)components. In addition KCB performs vibration measurements, which showed that both MCPs are in excellent conditions in terms of their vibration behavior. The MCPs have only been fully dismantled during a few occasions. In 1977 and 1984 MCP 1 was opened without disassembly of impeller and diffuser, the internal surfaces were inspected and roughness measurements were conducted without any findings. In 1989 MCP 2 was opened, no significant degradation was found.

Based on the above discussed measures, in combination with the results of in-service inspections, properly documented in inspection reports and annual ISI reports, and the extensive information in the MCP and primary supports AMR [15] [10], it can be concluded that the current physical status of the MCP (sub) components including their supports is within the predicted scope of the design.

2.3.5 Current physical status of the control rod drive mechanisms housings

In chapter 5 of the CRDM pressure housings AMR [16] relevant information to assess the current physical status of the CRDM pressure housings is provided. The highlights of the CRDM pressure housings relevant modifications, operation and maintenance history, including results of ISI, and the external operating experience and research are presented below:

According to available documentation, no modifications have been performed on the KCB CRDM pressure housings since commissioning of the plant.

The CRDM pressure housings are not periodically maintained or inspected. Up to now KCB did not experience any problems related to the CRDM pressure housings. External experience with CRDM pressure housings of similar design and materials, which have been in operation for a maximum time period of 38 years, also reports no problems up to now.

Based on these highlights and the extensive information in chapter 5 of the CRDM pressure housings AMR report [16] alone, no explicit statement on the current physical status of the CRDM pressure housings can be made. However, the outcome of the AMR assessment, concluded that no known ageing mechanisms are expected to have degraded the CRDM pressure housings significantly. In order to detect any signs of corrosive attack and to determine the current status of the CRDM pressure housings the AMR recommended to extend the ISI activities (see section 3.9.4).

2.3.6 Current physical status of the MCL and surge line

In chapter 5 of the MCL and SL AMR [17] relevant information to assess the current physical status of the MCL and SL is provided. The highlights of the MCL and SL relevant modifications, operation and maintenance history, including results of ISI, and the external operating experience and research are presented below:

Since the construction and commissioning phase some small modifications have been performed on the MCL system. The seals for the devices, on the hot and cold legs in both loops that were intended to allow isolation of the RPV from the remainder of the primary circuit during the initial design phase (Blasenstutzen), were changed twice. In 1985, the original graphite rings used to seal the Blasenstutzen connection were replaced by Kammprofile gaskets with graphite overlay. Leakage occurred in 1986 at this joint due to a wrongly adapted tightening procedure to fulfill ALARA requirements. In 1987, the Kammprofile gasket configuration was removed, and the Blasenstutzen were closed permanently through the installation of C-Torus seals.
Another modification was the addition of several TA-valves in 1979 to ensure that a second valve is installed at the safety class 1 to safety class 2 break. This modification made the isolation between safety class 1 and safety class 2 systems compatible with the class-break isolation as used in ANSI/ANS standards and therefore reduced the number of required in-service testing and inspections in the TA-system.

The design calculations of the MCLs and SL include the dimensioning of the components, elasticity calculations, stress analysis and fatigue analysis. In order to assure that the design assumptions made in these calculations are still valid, KCB regularly evaluates the actual number of transients to assure that these are below the anticipated number of transients. The actual number of transients important to the MCL and SL is still well below the anticipated number of transients assumed during the design phase. In addition to these design calculations a stress and displacement analysis for the TA nozzles of the MCL was performed in 1985. A leak before break (LBB) analysis was performed in 1990 to demonstrate that the MCLs and SL comply with the fundamental requirements for break preclusion. In order to better understand the occurrence of stratification in the surge line at KCB, thermocouples were placed at six locations on the SL to determine the extent of thermal stratification. This analysis recommended to remain in the hot stand-by state for only limited periods of time. This recommendation is implemented in the current operating procedures.

The MCL and SL (sub)components are periodically inspected using UT, PT and VT. During the last 10 years (2001-2010) no unacceptable indications were detected for in-scope MCL and SL (sub)components. Regular maintenance of the MCL and SL (sub)components is restricted to the seals. Besides the regular activities listed in the preventive maintenance, surveillance and/or ISI-programs, KCB performed several additional inspections. These included a one-time inspection of the welds in the MCL and SL, that are not inspected during regular inspections, as part of the LBB demonstration and additional visual inspections on both MCL loops using a robotic submarine (SUSI).

Based on the above discussed measures, in combination with the results of in-service inspections, properly documented in inspection reports and annual ISI reports, and the extensive information in the MCL and SL AMR [17], it can be concluded that the current physical status of the MCL and SL is within the predicted scope of the design.

2.3.7 Current physical status of the steel containment

In chapter 5 of the steel containment AMR [18] relevant information to assess the current physical status of steel containment is provided. The highlights of the steel containment relevant modifications, operation and maintenance history, including results of ISI, and the external operating experience and research are presented below:

The steel containment did not undergo any modification after construction of the plant.

The design analyses of the containment considered different load cases including earthquakes, aircraft crash, and loss of coolant accident (LOCA) following a main pipe break setting free a maximum of energy and mass. The highest load the steel containments has suffered is its pressure test during commissioning. Other loads on the steel containment are the integrated leak rate tests. Until now KCB performed these tests 7 times. Considering the test pressure of 2 bar and the test frequency, the load on the steel containment can be assumed to be negligible with regard to fatigue.

The steel containment is periodically visually inspected. No significant indications of the steel containment have been identified since the operation of the plant. Leakages in the frame of the airlocks have been detected during a leak tightness test. These leakages were attributed to failing seals. No further leakages were observed after their replacement. Based on international experience with respect to leaks of the steel containment and the good results from the previous integrated leak tests, the test interval was in agreement with the regulator expanded from 4 to 10 years in 1998.

Based on the above discussed measures, in combination with the results of in-service inspections, properly documented in inspection reports and annual ISI reports, and the extensive information in the steel containment AMR [18], it can be concluded that the current physical status of the steel containment is within the predicted scope of the design.

2.3.8 Current physical status of nuclear safety systems

In chapter 5 of the nuclear safety systems AMR [19] relevant information to assess the current physical status of the nuclear safety system SCs is provided. The highlights of the nuclear safety systems relevant modifications, operation and maintenance history, including results of ISI, and the external operating experience and research are presented below:

Since the construction and commissioning phase several modifications have been performed at the nuclear safety systems. Part of the nuclear safety system modifications resulted from periodic safety reviews. Before the first 10-yearly safety review, building 33 was erected. This building houses the, at that time, new installed emergency diesels (EY040/050), RS and TW-system. The first 10-yearly safety review resulted in the project “Modificaties” of 1997. This project implemented several safety measures related to nuclear safety systems, which included amongst others:

- Addition of the ultimate heat sink VE (chain TG080/TE/VE) which exclusively fulfills safety related tasks during beyond design accidents.
- Enlargement of diesel power by the installation of emergency diesel generators (EY010/020/030).

During the second 10-yearly safety review a number of further safety optimizations were identified which were implemented in the project “Mod2Go “in 2004. Modifications on nuclear safety systems included amongst others:

- Larger diesel fuel stocks for emergency diesel generators (EY).

Besides the modifications resulting from the periodic safety reviews, modifications on nuclear safety systems due to evaluations of maintenance, surveillance and ISI activities, as well as internal and external operating experience have been implemented. These included for example:

- Replacement of the coating of the VF-piping by IRATHANE®155 to improve the resistance of the coating against accidental oil spills on the Schelde river.
- Modifications of the reversing cover of the TF heat exchanger using modern CFD calculations, to mitigate fouling of the tube sheet by equalizing the flow distribution.
- Enlargement of the reversing head length of the EY030 heat exchanger and changing the tube material from CuNiFe10-90 to CuNiFe70-30, to prevent leakage of the heat exchanger tubes. Leakage of these tubes has occurred at the entrance side of the water in the past. This was caused by a too high velocity of the water, preventing the formation of a protective oxide layer.

Evaluation of maintenance, surveillance and ISI activities, as well as internal and external operating experience also resulted in improved programs and procedures. Based on historical maintenance information, it was for example decided to refurbish the VF-pump housings only every 8 years, mandated by the wear rate of its (glass flaked) coating.

Inspections, maintenance and surveillance activities have not identified any significant further degradation effects in the last 20 years. The majority of the maintenance activities are pro-active, such as mitigating the risk of chloride-induced corrosion by replacing all possible chloride containing gaskets and stuffing boxes by “chloride free” gaskets and stuffing box materials. Furthermore all consumables are marked for their use in and near systems to mitigate the influence of detrimental materials and reducing degradation. KCB also introduced procedures to reduce the introduction of foreign materials and objects into their systems.

Based on the above discussed measures, in combination with the results of in-service inspections, properly documented in inspection reports and annual ISI reports, and the extensive information in the nuclear safety systems AMR [19][25], it can be concluded that the current physical status of the nuclear safety system SCs is within the predicted scope of the design.

2.3.9 Current physical status of secondary systems

In chapter 5 of the secondary systems AMR [20] relevant information to assess the current physical status of the secondary system SCs is provided. The highlights of the secondary systems relevant modifications, operation and maintenance history, including results of ISI, and the external operating experience and research are presented below:

Since the construction and commissioning phase several modifications have been performed at the secondary systems. Part of the modifications resulted from periodic safety review measures. The safety measures of the first periodic safety review were implemented during the project “Modificaties” in 1997. These included amongst others:

- Several modifications to the main steam system (RA) e.g.,
    - Replacement of main steam lines inside the containment, to qualify them as Leak Before Break (LBB).
    - Introduction of multiple safety relief valves to obtain more redundancy.
    - Introduction of a flow restrictor in the main steam lines to limit the consequence of a line break.
- Several modifications to the emergency and main feedwater system (RL)e.g.,
    - Replacement of feedwater lines inside containment, to qualify them as LBB.
- Modifications to the back-up emergency feedwater system (RS) to develop a two-loop system with a capacity of 2 x 100 % with respect to the removal of residual heat.
- The VG pool and RZ pool liners were modified by applying PE liners.

During the second 10-yearly safety review a number of further safety optimizations were identified. These optimizations were implemented in the project “Mod2Go” in 2004 and included amongst others:

- Addition of a connection possibility from the back-up emergency feedwater system (RS) to a mobile fire water pump outside building 33 during accident conditions.
- The addition of an interconnection line between the two back-up emergency feedwater tanks, allowing the available back-up emergency feedwater pump to take suction from both RS tanks, and thereby extending the independence of the RS-system.

Besides the modifications resulting from the periodic safety reviews, several additional modification have been implemented. These included for example:

- Replacement of the moisture separators RB001/RB002 B001to improve the water separation efficiency and increase the thermal efficiency of the plant.
- Upgrade of the High Pressure (HP) turbine and Low Pressure (LP) turbines, including turbine inner housing and related parts, to increase the thermal efficiency of the plant.

Evaluation of maintenance, surveillance and ISI activities, as well as internal and external operating experience resulted in improved programs and procedures. Flow Accelerated Corrosion (FAC) was the dominant degradation mechanism in the early years until 1986. In the early years several system parts needed replacement because of FAC damage. These included the cold reheat line (RB), the turbine extraction lines (RF), the drain lines of the turbine piping (SH), parts of the extraction or condensate lines of RP, RM, RH, RG and several heat exchanger bundles. Furthermore parts were locally repaired or protected by applying a Metcoloy coating. In the eighties KCB switched from phosphate chemistry for the secondary side water chemistry to High All Volatile Treatment (HAVT). The most important influenceable parameter for FAC is the pH, which changed from 9.1 to 9.3 in the period 1976 to 1981 and from 9.3 to 9.8 in the period 1981 to 1986. From 1986 onwards, the pH is kept at a value ≥ 9,8. By the introduction of HAVT, the FAC rate was reduced significantly. This was, and still is, verified by the FAC measurements throughout KCBs systems. Based on ongoing knowledge and experiences the original FAC program was evaluated and further improved in 1988, 2004 and 2010.

No further, significant findings were identified during recent maintenance and ISI activities.

Based on the above discussed measures, in combination with the results of in-service inspections, properly documented in inspection reports and annual ISI reports, and the extensive information in the secondary systems AMR [20] [25], it can be concluded that the current physical status of the secondary system SCs is within the predicted scope of the design.

2.3.10 Current physical status of safety related auxiliary systems

In chapter 5 of the safety related auxiliary systems AMR [21] relevant information to assess the current physical status of the safety related auxiliary system SCs is provided. The highlights of the safety related auxiliary systems relevant modifications, operation and maintenance history, including results of ISI, and the external operating experience and research are presented below:

Evaluations of maintenance, surveillance and ISI activities, as well as internal and external operating experience resulted in several modifications. These included for example:

- Replacement of all cam profile gaskets with IT400 layer with Graphite layer gaskets.
- Improvements at the nozzles and manhole of both boric acid storage tanks (TB) were made using reinforcement rings, due to leakage at the manway gaskets.
- Repair and partly replacement of parts of the TA pressure reduction control valves due to degradation by cavitation. After a process change within design limits and straight replacement of some inner parts, cavitation appeared to occur, causing damage. A new design of control valve is in progress for future implementation, which should eliminate cavitation without process limitations.

Since 1982 KCB provides extra emphasis on the use of halogen low or halogen free materials. This resulted in implementing a system for condition based use of consumables and a complete restriction on some products containing halogens. The emphasis was also on asbestos materials, in which chlorides cannot be excluded. These asbestos materials were replaced by graphite materials or other low or halogen free materials.

In response to external operating experience KCB inspected a large sample of so called “Bretschneider-verschlusse” valves and flanges to identify corrosion. These inspections were without any significant result. Nowadays relevant valves and flanges are inspected when they are opened for maintenance and the inspection results are documented.

No further significant findings were identified during recent maintenance and ISI activities.

Based on the above discussed measures, in combination with the results of in-service inspections, properly documented in inspection reports and annual ISI reports, and the extensive information in the safety related auxiliary systems AMR [21] [25], it can be concluded that the current physical status of the safety related auxiliary system SCs is within the predicted scope of the design.

2.3.11 Current physical status of HVAC systems

In chapter 5 of the HVAC systems AMR [22] relevant information to assess the current physical status of the HVAC SCs is provided. The highlights of the HVAC systems relevant modifications, operation and maintenance history, including results of ISI, and the external operating experience and research are presented below:

Since the construction and commissioning phase several modifications have been performed at the HVAC systems. These included for example:

- Replacement of the three Freon 11 UV30, UV031 and UV032 chillers for thee new ones after corrosion in on the freon side in the early eighties.
- Installation of backup air cooled chillers UV033DS001/D002 in the early eighties.
- Replacement of coolant in the YV030, UV 031 and UV033 chillers in 1992. The freon 11 coolant was replaced with SUVA 123.
- Replacement of the air inlet filters with a new type of filters. These filters have an improved resistance against fog conditions and improved filter lifetime.

As a result of the first 10-yearly safety review the following modifications were included in project “Modificaties”:

- Increased capacity and improved filtering of the inlet air systems UW021 and UW22.
- Replacement of the three containment valves per line of TL04, TL10 and TL75 by two more reliable containment valves with leak test provisions per line.
- Installation of a new recirculation air cooling system for building 72 (emergency diesel EY010 and EY020).

During the second 10-yearly safety review a number of further safety optimizations were identified which were implemented in the project “Mod2Go” in 2004. Modifications on HVAC systems included:

- Replacement of the air inlet check valve TL070S004 to increase the reliability of the component.
- Addition of extra air coolers in the TL040 system to comply with the reduced allowable air temperature requirements for continuous work in the containment.

Besides the modifications resulting from the periodic safety reviews, KCB proposed several modifications on HVAC components for LTO. These include:

- TL000G001 intake air filter;
- TL00B002 inlet air heater;
- TL004 inlet air heater to building 01;
- TL006 inlet air heater to building 03;
- TL007 inlet air heater to building 03 laboratorium area;
- TL008 inlet air heater to building 03 office area;

- Replacement of the UV-chillers. These are planned to be replaced as the used refrigerant will be abandoned in 2014.

During recent surveillance activities a deficiency in piping support (UV006) was identified. It appeared that local damage of cold insulation had led to moisture condensation along the support and corrosion of the support and local pipe section. This pipe, its support and the insulation have been replaced.

Based on the above discussed measures, in combination with the results of in-service inspections, properly documented in inspection reports and annual ISI reports, and the extensive information in of the HVAC systems AMR [22] [25], it can be concluded that the current physical status of the HVAC SSCs is within the predicted scope of the design, when the proposed modifications for LTO are implemented.

2.3.12 Current physical status of RPV internals

In chapter 5 of the RPV internals AMR [23] relevant information to assess the current physical status of the RPV internals is provided. The highlights of the RPV internals relevant modifications, operation and maintenance history, including results of ISI, and the external operating experience and research are presented below:

The RPV internals are periodically inspected, mainly using UT and VT. These inspections resulted in several modifications during the service life of the plant. These included:

- Replacement of the baffle-former bolts, excluding the bolts in the lowest row, by austenitic stainless steel “star bolts”
- Replacement of the original disc springs (1.4122) of the hold-down assembly by commonly used alloy X-750 disc springs.
- Replacement of the original schemel bolts as a consequence of primary water stress corrosion cracking (PWSCC). PWSCC of the new installed bolts has been reduced due e.g., improved heat treatment techniques during manufacturing.
- Replacement to the fuel assembly centering pins.

The baffle-former bolts were inspected VT and UT. Based on small deviations in the UT signal (still acceptable result) a number of bolts have been replaced. A number of bolts were replaced preventively and a small number of bolts were replaced based on a visual crack in the intentionally deformed bold locking cup. Except for these baffle-former bolts, no further indications were found during the last 10 years (2001 – 2010).

Based on the above discussed measures, in combination with the results of in-service inspections, properly documented in inspection reports and annual ISI reports, and the extensive information in the RPV internals AMR [23] it can be concluded that the current physical status of the RPV internals is within the predicted scope of the design.

2.3.13 Current physical status of electrical and I&C systems

Since the construction and commissioning phase several modifications, e.g., bunkered back-up system (building 33), project “Modificaties” and “Mod2Go” have been performed. These modifications resulted in extension and/or replacement of the existing electrical and I&C SCs. The service life of a considerable part of the in-scope SCs is therefore far away from the assumed design life (assumed to be 40 years, however the maximum service life strongly depends upon the environmental conditions).

The AMR for electrical and I&C systems includes a plant walk down, interviews of KCB employees, as well as an evaluation of KCBs operating experience databases (VOB and SWG). During the plant walk down all areas in buildings 02, 03, 04, 05, 10, 21, 33, 35, 72 were visually examined. The areas in building 01 were extensively investigated in the EQDBA part of the project “LTO Bewijsvoering”. The plant walk down concluded that in general the visual condition of the electrical and I&C components in these areas was good. However some components, mainly cable trays, showed corrosion indications and some components were exposed to wetted conditions [42]. The interviews, as well as the evaluation of operating experience showed that KCB experienced some problems with electrical an I&C commodities in the past. These include for example insulation embrittlement, hardening and/or discoloration of wires, especially within the I&C cabinets and spreader. A large part of the spreader wiring was replaced in 1997. Nowadays KCB has a monitoring program for the spreader wiring in place and the wires in the I&C cabinets are regularly visually inspected. Other experienced, and fixed, problems are related to the commodity connectors and terminals and include amongst others damage of the swivel connection in junction boxes. However the general conclusion of the interviews with KCB employees, as well as the evaluation of operating history revealed that the electrical and I&C commodities are in a good condition.

Based on the results of the plant walk down, in combination with the interviews and the evaluation of operating history performed during the electrical and I&C AMR [27], it can be concluded that the current physical status of the in-scope electrical and I&C commodities is within the predicted scope of the design.

2.3.14 Current physical status of structures

The AMR for structures included a plant walk down, as well as an evaluation of KCBs maintenance and operating experience. These activities resulted in the following highlights:

- During an inspection in 1988 degradation of the reinforced concrete wall of the ventilation stack (part of building 03) was identified. The damaged concrete was repaired. Since that time the ventilation stack is protected by a three layered coating, whose top layer protects as a moisture repellant. The reinforced concrete and its coating has been inspected regularly. These inspections showed that the constructive protection against external influences is intact. However the coating showed cracks in some locations and the underlying concrete wasn't good everywhere. Therefore, KCB started a 2nd repair program in 2008, which was finished in 2009.
- In 1998 a break of the main anchors of the turbine building (04) steel structure was identified. Evaluation revealed that a too high pre-stress of the anchors and inappropriate storage and treatment of the anchors during the construction phase caused this problem. The anchors were replaced and these anchors are regularly visually inspected.
- Due to some damage of the sump coating it was in 1999 determined that the top cement layer of the floors in some installations rooms (building 01) needed to be replaced. KCB applied a new reinforced top layer. Nowadays the coating is inspected during each outage and coating damages are repaired immediately.
- The concrete of the cooling water intake and outlet building (21/23) and steel coating of the cooling water piping demand repair work, which is scheduled for 2011/2012.

Based on the above discussed measures, in combination with the results of inspections and maintenance activities, as well as the results of the plant walk downs and the extensive information in AMR report for structural components [26], it can be concluded that the current physical status of the in-scope structures is within the predicted scope of the design.

3 Results of the mechanical AMR

This chapter describes the results of the mechanical AMR. For each KCB relevant ageing mechanism a condensed description and the mechanical SCs of concern, as identified in the AMR, are provided. Thereafter the ageing management activities that serve to manage the ageing mechanism during LTO are evaluated. In case that the AMR identified that the existing ageing management activities are not adequate or do not exist at KCB, the recommendations from the AMRs are provided in this chapter as well. The implementation of these recommendations is described in chapter 6.

3.1 Management of thermal ageing

Thermal ageing is generally characterized by thermally activated movements of atoms in a material during extended periods of time at elevated temperatures. Thermal ageing may occur with or without external mechanical loads. Thermal ageing processes can occur within a broad temperature range, depending on the environmental and material parameters.

Thermal ageing, which causes a loss of ductility and a decrease of the impact strength and fracture toughness of a material, may affect both low-alloy steel and cast stainless steel containing residual δ-ferrite. It can occur at temperatures beyond 250 °C. Cr, CrNi- or CrNiMo-type austenitic stainless steel (cast steel or weld metal) with significant amounts of residual δ-ferrite (> 15 vol.-%), austenitic-ferritic stainless steel and ferritic-martensitic stainless steel may be susceptible to thermal ageing. Austenitic stainless steel metal welds with a δ-ferrite content > 10 vol. % may also be susceptible to thermal ageing [7].

The basic method to prevent thermal ageing is to limit the δ-ferrite content through appropriate material selection and qualified welding procedures. Components which are already installed and are subject to applicable conditions (see Table 3-1) should be carefully examined during plant operation. The AMR results of these components are described below.

Pressurizer

For the martensitic stainless steel nuts, washers and restricting orifices in the pressurizer vent line/SEBIM valves, degradation of mechanical properties due to thermal ageing cannot be fully ruled out. However, even in the postulated case of degradation, any loss of mechanical integrity of the pressure boundary is not expected, based on the function and mechanical load of these parts. The restricting orifices are periodically replaced, which is adequate to prevent thermal ageing. It is recommended in the AMR that KCB assesses the possible ageing of relevant nuts and washers and considers replacement, if necessary, during periodic inspections or when a joint is opened to perform maintenance activities [13].

Steam generator

The ferritic-martensitic stainless steel comb structure screws and primary manway sealing plates may be susceptible to thermal ageing as the operating temperature for these components is above 250 °C. The U-tube bundles (including the comb structure) on the secondary side of the steam generators are visually inspected (VT-1) every 7 years and the primary manway sealing plates are replaced when the primary side of the steam generator is opened for Eddy current testing (which means they are replaced every 3 years). Even in the event that these plates were to age in the 3 year interim period, leakage of the sealing plates can be detected during the system leakage testing walkdowns (VT-2) that take place following each outage prior to plant start-up or through the leakage monitoring system. These measures are considered to be sufficient for adequate management of thermal ageing of the comb structure screws and primary manway sealing plates [14].

Secondary systems

The ferritic-martensitic stainless steel branched small-bore piping (between RA002S034 and RA002S035 and the connected valve body RA002S035) may be susceptible to thermal ageing as the operating temperatures for these components are higher than 250 °C. However the consequences of leakage or failure of this piping or valve generally has no effect on nuclear safety. The existing performance of regular plant walkdowns aimed at identifying leakages is sufficient to adequately manage thermal ageing of these components [20].

Table 3-1 AMR results for the components identified as susceptible for thermal ageing

(Sub)components	**Recommendations**
Pressurizer	
Martensitic stainless steel nuts, washers and the restricting orifices in the pressurizer vent line, as well as the nuts associated with the SEBIM valves.	Assess possible ageing of these components and consider replacement.
Steam generators	
- Ferritic-martensitic stainless steel primary manway	-

(Sub)components	Recommendations
sealing plates; - Ferritic-martensitic stainless steel comb structure screws in the secondary side tube supports.	
Secondary systems	
Ferritic-martensitic stainless steel branched small-bore piping between RA002S034 and RA002S035 and the connected valve body RA002S035.	-

3.2 Management of neutron induced ageing (irradiation embrittlement)

Irradiation embrittlement is a relevant ageing mechanism for certain areas within the RPV and the RPV internals. The degree of irradiation embrittlement depends on the material composition, irradiation, temperature, fast neutron flux and time of exposure [7]. The AMR results of the RPV and RPV internals are described below.

Reactor pressure vessel

As the highest neutron fluence occurs in the core beltline region of the RPV, irradiation-induced material embrittlement is expected to occur within this region, especially within the weld joints (see Table 3-2). A safety assessment, with respect to irradiation embrittlement of the RPV considering 60 years of operation has been performed. This assessment shows that sufficient safety margin exists and that the calculated allowable reference temperature with respect to nil-ductility transition is within the existing limits according to KTA rules. In addition, KCB will validate the expected low level of irradiation embrittlement of the RPV through implementation of an additional RPV irradiation surveillance program during LTO. The use of low-leakage core management results in further limitation of neutron and gamma irradiation levels in the RPV core beltline region. Irradiation embrittlement of the RPV is adequately managed with these ageing management activities in place at KCB [11].

Reactor pressure vessel internals

Areas of the RPV internals (including the core internals) composed of austenitic stainless steel and nickel-base alloys are subject to irradiation embrittlement (see Table 3-2). As the fluence thresholds for both irradiation embrittlement and Irradiation Assisted Stress Corrosion Cracking (IASCC) are close together, the inspection activities performed for IASCC are sufficient to envelop irradiation embrittlement as well (see section 3.7.4). Therefore, the ageing management for neutron-induced ageing of the RPV internals at KCB is adequately covered by the activities in place for the management of IASCC [23].

Table 3-2 AMR results for the components identified as susceptible for irradiation embrittlement

(Sub)components	Recommendations
RPV	
RPV core belt region	-
RPV internals	
- Core baffle; - Core barrel; - Former plates; - Baffle-former bolts; - Barrel-former bolts; - Upper and lower fuel assembly centering pins; - Lower core support; - Grid plate; - Bolts of in-core instrumentation tube.	-

3.3 Management of ageing by frictional impact (wear)

Wear is defined as the loss of material from interacting surfaces under relative motion, resulting in a tribological process. Wear occurs as a consequence of friction and is promoted in the presence of frictional forces, surface roughness, abrasive particles and corrosive fluids. In general, materials with high strength and toughness, such as certain stainless steel and nickel alloys show increased resistance to wear [7].

Most in-scope structures and components are generally immovable parts and are, therefore, not susceptible to ageing by frictional impact. The structures and components for which ageing by frictional impact has been identified as a relevant ageing mechanism are listed in Table 3-3.The AMR results of these SCs are described below.

Steam generator

Wear of the steam generator U-tubes has been identified to occur as a consequence of tube support failure. Nowadays the tube supports incorporate special comb-shaped constructions in the U-bend region, restoring the original tube support and minimizing flow induced vibrations and fretting. However, even considering this improvement, degradation by frictional impact due to support failure cannot be completely excluded. Loose parts which could accumulate in low flow areas within the primary and secondary side of the SGs can also cause damage to the rolled tube plugs and U-tubes.

<u>Primary side</u>

KCB can detect loose parts which can lead to damage due to frictional impact, using the loose parts monitoring (KÜS) system. However only the larger loose parts that are present in the primary and secondary sides can be identified using this system. KCB adheres to a foreign material exclusion procedure to prevent the introduction of external materials (e.g., brought in through inspection or maintenance activities by personnel) in the primary system. In the case wear / degradation (wall thinning) at the U-bundle or near the tubesheet has occurred, this can be detected during regular Eddy current testing. During the last inspection period KCB also inspected the rolled tube plugs. However this inspection is not part of the ISI plan at the moment. It is therefore recommended in the SG AMR that this inspection is incorporated into the ISI plan, coinciding with other primary side SG activities [14].

<u>Secondary side</u>

The U-tube bundle on the secondary side of the steam generator is visually inspected (VT-1) every 7 years. This inspection includes visual inspection of the tube supports to detect degradation / wear of secondary subcomponents. Additionally the integrity of the U-tubes is investigated through 3-yearly Eddy current and ultrasonic testing. The performance of these inspections complement each other in serving to identify the onset of wear in the area of the U-tube bundle supports and tubesheet at an early stage.

In 2004 foreign objects and deposits were identified by visual inspections after chemical cleaning of the secondary side of both SGs. These foreign objects could damage the U-tubes due to fretting. AREVA NP made an evaluation of these foreign objects under the assumption of NPP operation until 2013. To ensure that these objects are still in position it is recommended in the SG AMR to perform a one-time inspection on the top of the tubesheet (VT-1) and determine if any present objects can remain in place throughout LTO. With the performance of this additional inspection ageing by frictional impact is adequately managed for the secondary side of the steam generator [14].

Main coolant pump

The slide rings of the MCP HP and LP shaft seals are susceptible to ageing by frictional impact, which could lead to large amounts of seal leakoff. However KCB continuously monitors the flow rate of the HP and LP seal leakoff lines. If the flow rate in the LP seal leakoff line becomes too high, the MCP is taken out of service and the shaft seals are maintained (e.g., repaired). Ageing of the slide rings due to frictional impact is adequately managed by the performance of these activities [15].

Reactor pressure vessel internals

The bolts and pins on removable parts, as well as alignment and interface components of the RPV internals are susceptible to ageing by frictional impact. To detect broken parts KCB has installed the loose part monitoring (KÜS) system. Additionally KCB inspects annually the alignment of the grid plate (YH101/205) and the upper and lower fuel assembly centering pins (YH110/206). Each refueling cycle KCB performs a general test for checking wear at components of the RPV internals using a functional test of the control rod guide assemblies (CRGA) (movability). With the performance of these activities ageing by frictional impact of the RPV internals is adequately managed [23].

Supports and hangers

Piping vibration can affect pipe supports by causing wear between the component (e.g., piping) and its support (e.g., pipe clamp). Thermal expansion of piping may also affect the interface between piping and its supports. KCB performs several plant walkdowns (e.g., system leakage testing walkdowns (after each outage), operator plant walkdowns (1x per shift), multidisciplinary walkdowns and maintenance specific walkdowns). The performance of visual inspections through these walkdowns identifies degraded supports or hangers during plant operation. Therefore, it is concluded that ageing by frictional impact is adequately managed [24].

Table 3-3 AMR results for the components identified as susceptible for frictional impact

(Sub)components	**Recommendations**
Steam generator	
- Tube plugs (rolled)	Add a visual inspection of the rolled tube plugs to the ISI plan.
- U-tubes	Perform an one time VT-1 of SG tubesheet and evaluate if foreign objects can remain in place throughout LTO.
Main coolant pump	
HP and LP shaft seal slide rings	-
RPV internals	

(Sub)components	Recommendations
Bolts and pins on removable parts, as well as alignment and interface components, these include: - Pins for control rod guide assembly; - Upper and lower fuel assembly centering pins; - Alignment of grid plate.	-
Supports and hangers	
Surfaces of all supports and hangers	-

3.4 Management of ageing at elevated temperatures with applied stress and strain (creep and relaxation)

Creep and relaxation are both time and temperature dependent degradation mechanisms.

3.4.1 Creep

Creep is defined as time and load dependent plastic deformation of a material exposed to an applied stress. Metallic materials are generally susceptible to creep when the operating temperature exceeds approximately 40 % of the materials absolute melting temperature (in degrees Kelvin). Taking into account the intended service temperatures under normal operating conditions at KCB and the neutron irradiation dose to which components are subjected, the risk of creep damage is not relevant for any in-scope component materials. Therefore, creep can be classified as an insignificant degradation mechanism for all in-scope components [7].

3.4.2 Relaxation

A majority of the stress relaxation of a joint occurs directly after tensioning of the flange. The amount of stress relaxation depends highly on the applied bolt material, the tensioning procedure and the configuration of the joint. Stress relaxation could lead to untightening and may result in leakage of the relevant flange connections [7]. Leakage from relaxed fasteners is not a safety issue in terms of plant operation, but an asset management issue [2].

[2] Leakage from relaxed fasteners could cause corrosion-induced degradation, which is considered in the AMR and discussed in section 3.6.

In the nuclear industry, closure bolting is generally only replaced when inspections have determined that the bolting is degraded, or if a joint is opened to perform certain maintenance or repairs, or due to adverse conditions or leakage. Certain joints are periodically opened at KCB to perform maintenance. Therefore, the corresponding flange connections are periodically tightened and relaxation of these studs or bolts is managed. However certain joints within the scope of this AMR are not opened periodically. Given their location, some of them are subject to high temperatures (approximately 300 °C) during normal operation, relaxation of the relevant mechanical fasteners could theoretically occur (see Table 3-4). Even in the unlikely event that leakage of a flange connection occurs, it can be identified through visual inspections during plant walkdowns (e.g., system leakage testing walkdowns (after each outage), operator plant walkdowns (1x per shift), multidisciplinary walkdowns and maintenance specific walkdowns) and/or through the leakage monitoring system. In the case that leakage is detected through either plant walkdowns or the leakage monitoring system, maintenance or repair activities are conducted (regardless whether boron is present or not) to identify the cause of the leakage, as well as correct the condition, as necessary. With the performance of the above mentioned activities relaxation the mechanical fasteners listed in Table 3-4, excluding the fasteners of the RPV internals, is adequately managed at KCB [11] [13] [14] [15] [17] [24] [25]. The AMR results for the fasteners of the RPV internals are described below.

Reactor pressure vessel internals

Relaxation of RPV internals fasteners (see Table 3-4) is difficult to manage by surveillance activities, as the effects of relaxation cannot be detected by visual inspections. However the KCB ISI-plan includes the control of the form closure between the core baffle plates (YH106). This inspection is sufficient for detecting the effects of relaxation of the baffle-former and barrel-former bolts. Relaxation of the disc springs of the hold-down spring assembly is adequately managed at KCB by the periodic measurement of the spring constant of the disc springs (YH 211) [23].

Table 3-4 AMR results for the components identified as susceptible to relaxation

(Sub)components	**Recommendations**
RPV	
Fasteners of the reserve level measurement nozzle blind flange and center CRDM nozzle blind flange	-
Pressurizer	
- Connection flange fasteners;	-

(Sub)components	Recommendations
- Manway fasteners; - Heating element fasteners; - RPV fasteners; - Pressurizer vent line fasteners; - Bolts (M42) and reduced shank bolts associated with the SEBIM valve blocks.	
Steam generator	
- Secondary side manway and handhole cover studs; - Comb structure screws.	-
Main coolant pump	
- Fasteners on MCP cover; - Fasteners of the flange connections in the emergency seal water supply lines.	-
Main coolant lines	
Fasteners on the blasenstuzen	-
RPV internals	
- Baffle-former bolts; - Barrel-former bolts; - Disc springs hold-down spring assembly.	-
Supports and hangers	
Fasteners of supports and hangers	-
Mechanical fasteners	
Fasteners in nuclear safety systems, safety-related auxiliary systems, secondary systems and HVAC systems	-

3.5 Management of ageing under cyclic or transient loading (fatigue)

Metals that are subject to cyclic mechanical and/or thermal loading are generally susceptible to fatigue, depending on material properties and the presence of inhomogeneities, design and surface quality, residual stresses, loading conditions and the environment. To mitigate fatigue of metallic (sub)components, the application of sufficient design margins and minimization of thermal gradients is necessary. Areas of concern generally include (sub)components exposed to mixing zones of cold and hot water (risk for thermal stratification), as well as components subject to thermal shock and vibrational forces (see Table 3-4).

To mitigate the effects of ageing due to fatigue on component functions during LTO, KCB implemented fatigue management. Fatigue management is focused on the prevention of crack initiation and is achieved through the employment of a combination of analyses (e.g., stress analyses and subsequent fatigue analyses) and condition monitoring activities (e.g., fatigue monitoring system (FAMOS[3]) and periodic vibration measurements). For applicability during LTO KCBs existing fatigue analyses are revalidated to incorporate the extended operating period. The preliminary results for most in-scope components show Cumulative Usage Factor (CUF) values less than 1,0 during 60 years of service life. However the analyses for some (sub)components have not yet been completed. With the completion of the revalidation of existing fatigue analyses and the validation with FAMOS, fatigue for most in-scope components and subcomponents is adequately managed [11] [14] [20] [21] [25]. The AMR results of the remaining structures and components listed in Table 3-4 are described below.

Main coolant pump

Besides the completion of the revalidation of existing fatigue analyses for LTO (see Table 3-4 for areas of concern) and the use of FAMOS to validate the assumptions in these analyses it is recommended in the MCP AMR to perform additional fatigue analyses for mechanical fasteners and the auxiliary lines as explained below.

Fatigue damage in several KSB main coolant pumps in Siemens/KWU NPPs recently occurred due to thermal fatigue of the shaft seal casings. However, these incidents cannot be transferred directly to the Sulzer MCPs at KCB, as the construction in the region of these failures is not identical. At KCB, main coolant water is conducted into the HP cooling circuit through the emergency seal water supply line through the HP-cooler and cooled down before entering the pump seal (see Figure 3-1). In this way, the occurrence of thermal shock is prevented in the region of the MCP cover and the seal insert. However, a thermal transient could occur in the region where the main coolant water (295 ^{0}C) in the emergency seal water supply line (YD001/002 Z006) flows into the HP cooling circuit (YD001/002 Z003) in the event that the valve (YD001/002 S005) is opened (see Figure 3-1).

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[3] FAMOS monitors thermal loading, including stratification in several fatigue relevant locations on the primary circuit and several location on the main and emergency feedwater system piping.

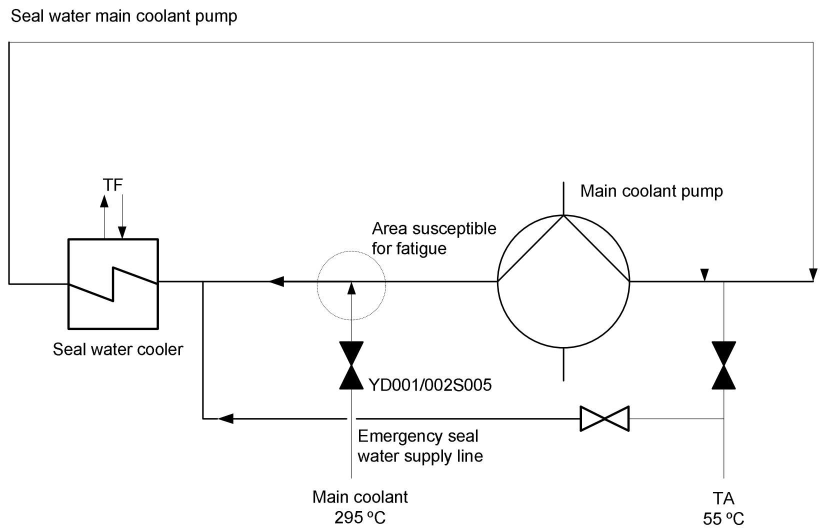

Figure 3-1 Schematic overview of seal water cycle of the main coolant pump

As the emergency seal water supply lines (YD001/002 Z003, Z005, S005) may be subjected to thermal transient conditions and no fatigue calculations were performed for the mechanical joints or the relevant auxiliary lines, an evaluation of these transients should be performed [15].

Main coolant lines and pressurizer surge line and Pressurizer

Besides the completion of the revalidation of existing fatigue analyses for LTO (see Table 3-4 for areas of concern) and the use of FAMOS to validate the assumptions in these analyses it is recommended in the MCL and PZR AMR to evaluate the current loading configurations during normal operation as explained below.

Thermal stratification is an applicable loading configuration for the surge line and its nozzles on the main coolant lines and on the pressurizer. Thermal stratification may occur, for example, in horizontal and slightly sloping parts of the surge line during start-up, as cooler and thus denser water from the primary circuit flows under the warmer and thus lighter water from the pressurizer. During start-up and shutdown the temperature difference in the surge line will be approximately 150-200 ^{0}C, while during normal power operation this difference is approximately 25 ^{0}C.

Thermal stratification should be minimized. Therefore, it is recommended in the MCL and PZR AMR that KCB re-evaluates the current loading configurations experienced during normal operating conditions, specifically start-up and shutdown, to determine if changes to certain operating procedures should be made to minimize the likelihood of temperature transient conditions (including specific limits for time period and ΔT between the MCL and surge line). This evaluation should also consider the design of the main coolant pump; to determine if the pump can start at a lower pressure [13] [17].

CRDM pressure housings

The design of the KCBs CRDMs did not require a fatigue analysis during the design phase due to low loading. Based on AREVA experience with the design of the CRDM pressure housing, it is also expected that fatigue should not be a major issue for this component. However, in the frame of ongoing EPR projects the scope of transient load specification has been extended and considerable thermal transient loads (due to the occurrence of SCRAM events or large stepping motions of the CRDM itself) should be evaluated to determine the actual fatigue loading on the CRDM pressure housings. These thermal loads are not specific EPR loads, they can also occur in the CRDMs of KCB as they are of similar design. Therefore it is recommended in the CRDM pressure housings AMR to evaluate if a fatigue analysis of the CRDM pressure housing is necessary for LTO [16].

Reactor pressure vessel internals

Fatigue is regarded as a relevant ageing mechanism for RPV internals bolts and pins (see Table 3-4). The possible occurrence of fatigue (e.g., cracking and fracture) of bolts and pins is adequately managed by KCB by the inspections (e.g., YH203; annual inspection of the CRGA bolts and YH212; inspection of the centering pins of the CRGA) listed in ISI- plan. In addition to the aforementioned inspections continuous monitoring for loosened or loose parts is in place using the loose parts monitoring system [23].

Supports and hangers

Vibrational fatigue can occur at piping supports. The effects on vibrational fatigue at supports (e.g., degraded supports or hangers) can easily be detected by visual inspections. With the performance of VT-3 inspections and visual inspections through plant walkdowns vibrational fatigue is adequately managed for supports and hangers [24].

Table 3-5 AMR results for the components identified as susceptible for fatigue

(Sub)components	Recommendations
RPV	
Relevant RPV subcomponents	Revalidate existing fatigue analyses.

(Sub)components	Recommendations
Pressurizer	
- Pressurizer spray lines (YP, TA, TW); - Pressurizer spray line nozzles; - Surge line nozzle at the pressurizer.	Revalidate existing fatigue analyses and validate assumptions using FAMOS.
- Surge line nozzle at the pressurizer	Re-evaluate current loading configurations during normal operation.
Steam generator	
- SG shell and cover; - Primary inlet and outlet nozzle; - Feedwater and main steam nozzle; - Tubesheet.	Revalidate existing fatigue analyses and validate assumptions using FAMOS.
Main coolant pump	
All in-scope YD system components and subcomponents (including the MCP cover fasteners and fasteners in the flange connections in the HP cooling circuit and HP and LP leakoff lines).	Revalidate existing fatigue analyses and validate assumptions using FAMOS.
Emergency seal water supply lines	Evaluate fatigue for the emergency seal water supply lines.
CRDM pressure housings	
Potentially all CRDM pressure housing parts	Evaluate if a fatigue analysis of the CRDM pressure housing is necessary for LTO.
Main coolant lines and pressurizer surge line	
All in-scope components and subcomponents (including fasteners)	Revalidate existing fatigue analyses and validate assumptions using FAMOS. Re-evaluate current loading configurations during normal operation.
RPV internals	
Bolts and pins: - Baffle-former bolts;	-

(Sub)components	Recommendations
- Former-baffle bolts; - Pins and bolts CRGS; - Bolts of in core instrumentation tube; - Schemel bolts.	
Secondary systems	
- Main feedwater system forgings (RL040/RL050 Z101); - Double T-junction nozzles (RL040/RL050 Z003 and RS011/RS021 Z003); - Main steam system piping forgings (RA001/RA002 Z102); - Main steam system piping forgings in building 04 (RA001/RA002Z002).	Revalidate existing fatigue analyses and validate assumptions using FAMOS.
Auxiliary systems	
- Tubesheets of the recuperative heat exchangers (TA000 B001/B002)	Revalidate existing fatigue analyses and validate assumptions using FAMOS.
Supports and hangers	
- Piping supports	-

3.6 Management of ageing without mechanical loading

3.6.1 General corrosion by dissolved oxygen

In high-temperature pressurized water (> 100 °C), in particular in Pressurized Water Reactor (PWR) primary and secondary circuit water, none of the normally applied construction materials are significantly susceptible to general corrosion based on the formation of stable protective oxides on the material surface. However general corrosion by dissolved oxygen could occur when unprotected surfaces of cast iron, carbon steel or low- alloy steel parts are subjected to aqueous solutions containing dissolved oxygen or to a moist environment. This may occur during:

- Outage conditions (cold) in high-temperature-water systems with access of air
- Start-up conditions at intermediate temperatures, as long as dissolved oxygen from outage conditions is not completely removed

- Moist operating conditions, e.g., when local temperatures are below local dew points
- In seawater (Schelde river) systems, the high content of chlorides will significantly enhance the corrosion of unprotected carbon steel and low-alloy steel surfaces [7].

Therefore, internal surfaces of cast iron, carbon steel, and low-alloy steel components may be susceptible to general corrosion by dissolved oxygen in aqueous environments at temperatures below 100 °C. External surfaces of cast iron, carbon steel, and low-alloy steel components are susceptible to general corrosion in humid conditions at temperatures below 100 °C. For conservatism also cast iron, carbon steel, and low-alloy steel coated components (e.g., rubber lined) are considered to be susceptible to general corrosion as a failure of the coating applied on different material surfaces cannot be excluded.

The zinc coating present on the surface of zinc galvanized steel components is also susceptible to general corrosion under certain conditions. In the pH range between 6-12, zinc undergoes a negligible amount of corrosion under moist environmental conditions. When exposed to aqueous environments outside of this pH range, the protective nature of the coating may be significantly reduced.

The water chemistry program at KCB mitigates general corrosion for internal surfaces of components exposed to water. The water chemistry program monitors the concentration of certain chemical species, including chlorides, fluorides, sulfates, dissolved oxygen and hydrogen and the chemistry properties are adjusted if needed. The chemistry parameters are monitored through the use of in-process methods or sampling. General corrosion of internal surfaces of components exposed to oil or diesel is mitigated through the quality analysis of the lubricating oil and/or diesel fuel. These programs monitor the contamination of lubricating oil and/or diesel fuel with water, dissolved oxygen and other impurities.

Besides these preventive activities KCB performs regular plant walkdowns (e.g. operator plant walkdowns (1x per shift) and multidisciplinary walkdowns) to identify degraded components in an early stage. Plant walkdowns will not identify degradation of heat exchanger internals. In general heat exchangers are regularly cleaned and visually inspected. In addition degradation of the heat exchanger tubes can be identified through variations in the heat exchanger outlet temperature.

With the above mentioned activities in place, it is concluded that general corrosion by dissolved oxygen is adequately managed at KCB for internal and external surfaces of all in-scope components [18][19][20][21][22][24][25].

3.6.2 Wastage

Wastage is a mode of general corrosion which is often associated with attack by phosphates in the secondary circuit of PWRs where phosphate treatment was formerly used [7]. Wastage is only a relevant ageing mechanism for the external surfaces of the steam generator U-tubes (see Table 3-6) [14].

KCB has used phosphate treatment in the secondary circuit previously. As this caused wastage (several tubes needed to be plugged) KCB switched to High All Volatile Treatment (H-AVT) in 1988. As a consequence, wastage is nowadays no longer of great concern. However, due to the use of phosphate treatment in former times, wastage at U-tubes can still occur in the area at the top of the tubesheet.

To mitigate the occurrence of wastage, it is necessary that the steam generators maintain a high level of cleanliness. Water chemistry measures are taken to minimize corrosion product transport and accumulation on the secondary side of the SGs. The material condition of the U-tubes at KCB are determined every 3 years using Eddy current testing. In addition Eddy current array testing (X-probe) and ultrasonic testing on the U-tubes is also applied. Eddy current testing can detect the presence and height of deposits that have accumulated on top of the tubesheet. It is recommended in the SG AMR to ensure that these measurements are part of the testing procedure. Based on the determination of expected deposit quantities KCB should determine the necessity for sludge removal from collection areas on the top of the tubesheet [14].

Table 3-6 AMR results for the components identified as susceptible to wastage

(Sub)components	**Recommendations**
Steam generator	
External surfaces of the U-tubes	Ensure that the presence and height of deposits that have accumulated on top of the tubesheet are measured and determine if sludge removal is needed.

3.6.3 Boric acid corrosion

Boric Acid Corrosion (BAC) is a form of uniform corrosion that attacks carbon steel and low-alloy steel in the presence of hot, concentrated aqueous solutions of boric acid. A considerable loss of material could occur in the case of continuous and long-term leakage of treated borated water at elevated temperatures on external carbon steel and low-alloy steel surfaces in an environment that contains oxygen. These conditions could for example exist near flange connections which contain treated borated water.

Boric acid corrosion is mitigated at KCB by minimizing primary coolant leakage through monitoring of the locations where potential leakage could occur (e.g., due to regular visual inspections of all accessible components of the primary circuit and adjacent systems and leakage monitoring), and subsequent corrective actions (e.g., cleaning or assessment of the affected components in the event that leakage occurs and timely repair).

With the activities in place, boric acid corrosion is adequately managed for all in-scope components and subcomponents [10][11][13] [14][15][17] [20][21] [24][25].

3.6.4 Pitting corrosion

Pitting corrosion is a form of localized corrosion attack of metallic surfaces in aqueous solution which are passivated by the surrounding environment. Typically, pitting occurs under conditions where a passivating protective oxide layer is damaged due to the chemical impact of specific ions. This allows aggressive ions, such as chlorides, to come into contact with the metal material surface.

Pitting corrosion of cast iron, carbon steel and low-alloy steel with passivated surfaces may be observed in high temperature water with increased levels of dissolved oxygen. Austenitic stainless steel may be susceptible to chloride-induced pitting in both the primary as well as the secondary side, in the (assumed) presence of chlorides. In systems with raw water, even austenitic stainless steel dedicated for high temperature-water systems becomes susceptible to chloride-induced pitting.

The susceptibility of nickel and nickel-base alloys is not expected in normal PWR temperature or chemistry conditions. Copper alloys in contact with an environment containing sulfates and a high concentration of chlorides can result in pitting corrosion. Copper is also susceptible to pitting corrosion when exposed to raw water at temperatures ≤ 40 °C and a pH between 6,5 and 7,5. Zinc galvanized steel could also be susceptible to pitting corrosion. In the pH range between 6-12, zinc galvanized steel is resistant to pitting corrosion because the sacrificial zinc layer protects the base metal even in the case that the zinc coating is damaged. However when zinc galvanised steel is exposed to aqueous environments outside of this pH range, the protective nature of the zinc coating may be significantly reduced.

Pitting corrosion is mitigated in the primary and secondary cooling systems at KCB as the main and secondary coolant has a sufficiently low corrosion potential and a constant pH of 7. The presence of high chloride concentrations is mitigated by implementation of the KCB water chemistry program. The chloride concentration during normal operation is below 0,005 ppm, which is far below the chloride concentration where pitting is likely to occur [7]. Additionally, the presence of chlorides (e.g., from gaskets or fitting lubricants), is nowadays managed at KCB through the use of chemicals or materials that have a low chloride content. Besides these preventive activities the external surfaces for all accessible components of the main coolant system are visually inspected during the leakage testing walkdowns (VT-2) following each outage prior to plant start-up. In this way the integrity of the pressure boundary is inspected and corrective actions are in place if adverse conditions were identified. The external surfaces of remaining systems are looked at during the regular plant walkdowns. These walkdowns are performed to identify in an early stage degraded components during plant operation. With these activities in place, it is concluded in the AMR that pitting corrosion is adequately managed for in-scope structures and components, [10] [11] [13] [15] [16] [17] [18] [19] [20] [21] [22] [23] [24] [25].

In order to adequately manage pitting corrosion for the SGs, it is recommended in the SG AMR [14] to determine the level of deposit accumulation on top of the tubesheet (see also section 3.6.2).

3.6.5 Intergranular corrosion / intergranular attack

Intergranular corrosion (IGC) is a selective intergranular attack of a metal concentrated at or adjacent to the grain boundaries. IGC may occur if a less noble or somewhat susceptible (i.e., sensitized) phase of a material or less noble alloying element is exposed to a corrosive environment. Austenitic stainless steel and nickel-base alloys, both otherwise corrosion resistant materials, may be susceptible to this form of corrosion.

Austenitic stainless steel may become susceptible to IGC due to thermal sensitization. The occurrence of IGC at PWRs is generally not expected since they operate at low oxygen concentrations and conditions are controlled during start-up [7]. To ensure that oxidizing conditions are indeed not present at KCB, the occurrence of IGC has conservatively been treated as a relevant ageing mechanism for thermally sensitized austenitic stainless steel components or subcomponents in the AMR [11] [15] [19] [20] [21] [22].

At KCB oxidizing conditions are mitigated, e.g., the oxygen content in the main coolant water is < 1 ppb during normal operation, through implementation of the water chemistry program. In addition, the application of chloride sources (e.g., gaskets or fitting lubricants) is mitigated at KCB through the use of chemicals or materials that have a low chloride content. Contamination of lubricating oil or diesel fuel with water, dissolved oxygen and other impurities is monitored through the quality analysis of the lubricating oil and/or diesel fuel. Besides these preventive activities KCB performs regular plant walkdowns to identify degraded components in an early stage during plant operation. With these activities in place it is concluded in the AMR that IGC is adequately managed for in-scope structures and components [11] [15] [19] [20] [21] [22].

In order to adequately manage IGC for the SGs, it is recommended in the SG AMR [14] to determine the level of deposit accumulation on top of the tubesheet (see also section 3.6.2).

3.6.6 Crevice corrosion

The term crevice corrosion is applied for corrosion which occurs in crevice conditions. This causes the enhancement of common corrosion mechanisms (e.g., pitting and general corrosion). Relevant cases for KCB include design-related crevices on carbon steel and low-alloy steel surfaces, in particular during outages when dissolved oxygen can enter the system. Austenitic stainless steel may become sensitive to crevice corrosion when chemical contaminants, for example chlorides, can accumulate during operation and outages [7]. If crevice conditions meet the conditions for pitting corrosion, crevice corrosion could also affect copper and copper alloys. The underlying steel base metal of galvanized steel could be susceptible to crevice corrosion when the protective zinc coating becomes degraded, which could occur depending on the conditions to which they are exposed (e.g., frequency and duration of exposure to moisture, pH, drying, and certain contaminants).

For most in-scope components and subcomponents design related crevices are not expected, but could not be ruled out completely. Only for the CRDM pressure housings and the primary components supports it was concluded that crevice corrosion does not need to be assessed in the AMR . Crevice corrosion of the CRDM pressure housings is not relevant as the dissimilar welds and the lower homogenous weld of the CRDM pressure housings were all machined after welding and welds were made in downhand position (position PA) rotating without any constraint [16]. Crevice corrosion of the primary supports can be ruled out as continuous and long term leakage in the only area with a possible accumulation of chemical contaminants (i.e. the RPV support construction below the reactor well lining) can be excluded (as the water on wetted surfaces evaporates after start-up) [10].

With respect to crevice corrosion of internal surfaces, the presence of high chloride concentrations and other impurities is generally mitigated by implementation of the KCB water chemistry program. In addition, the application of chloride sources (e.g., gaskets or fitting lubricants) is mitigated at KCB through the use of chemicals or materials that have a low chloride content. Contamination of lubricating oil and diesel fuel with water, dissolved oxygen and other impurities is monitored through the quality analysis of the lubricating oil and/or diesel fuel. Refrigerant analysis monitors and adjusts the chemistry properties of refrigerants. For example the quality of the refrigerant of the UV system is checked once or twice a year depending on the size of the cooling system [22].

Crevices corrosion is also managed via visual inspections (VT-1 and VT-3) of component surfaces (e.g., the seal surfaces of the RPV cover, the bearing surfaces of the core barrel and the bearing surfaces of the upper core support) performed during regular maintenance and inspection activities.

In the event that leakage would occur, it can be identified through visual inspections during plant walkdowns (the external surfaces for all accessible components of the main coolant system are visually inspected during the leakage testing walkdowns (VT-2) following each outage prior to plant start-up, while the external surfaces of remaining systems are looked at during the regular plant walkdowns) and/or through the leakage monitoring system. In the case that leakage is detected through either plant walkdowns or the leakage monitoring system, maintenance or repair activities are conducted to identify the cause of the leakage, as well as correct the condition, as necessary. With these activities in place it is concluded that crevice corrosion of most in-scope structures and components is adequately managed [11] [13] [15] [17] [19] [20] [21] [22] [23] [24] [25][26].

In order to adequately manage crevice corrosion for the SGs, it is recommended in the SG AMR [14] to determine the level of deposit accumulation on top of the tubesheet (see also section 3.6.2).

3.6.7 Galvanic corrosion

Galvanic corrosion is a phenomenon which may enhance general or localized corrosion that is currently in progress at a lower rate. Galvanic corrosion requires the metallic contact of two different metals with different Electrochemical Potentials (ECPs) immersed within an electrolytic solution. In high-temperature water environments, the ECPs of carbon steel, low-alloy steel, austenitic stainless steel and nickel-base alloys are almost equal and therefore do not cause galvanic corrosion at dissimilar metal interfaces in these conditions. At lower, ambient temperatures, however, the ECPs may be different. Consequently, dissimilar metal welds in low-temperature systems (i.e., ambient temperatures) are of concern with respect to galvanic corrosion [7].

Galvanic corrosion could theoretically occur in piping systems, but KCB experience shows that in practice the probability is very low. Comparatively small cathodes (i.e., the electrochemically more noble metal) cannot cause severe detrimental effects as the process is very slow under these circumstances, whereas large cathodes can be very detrimental when installed in combination with comparatively small anodes (i.e., the electrochemically less noble metal). In case galvanic corrosion would occur, leakages are detected by visual inspections through the performance of regular plant walkdowns. Based on these arguments it is concluded that galvanic corrosion is adequately managed for all in-scope subcomponents and components [18] [19] [20] [21] [22] [25].

3.6.8 Microbiologically Influenced Corrosion (MIC)

Microbiologically Influenced Corrosion (MIC) is a type of corrosive attack accelerated by the influence of microbiological activity. It is affected by the way these microbes change the local environment or local surface conditions. This local environmental change can enhance general corrosion or localized corrosion through pitting or IGC.

MIC is enhanced in clean natural waters with high biological activity. Favorable conditions are temperatures < 100 °C, intermediate pH (4-9), and locations with stagnant flow. MIC occurs preferentially in passive alloys, in particular austenitic stainless steel. However it may also occur in carbon steel and low-alloy steel [7].

MIC has been identified as a relevant ageing mechanism for several cast iron (rubber lined), carbon steel (rubber lined), stainless steel and copper (alloy) components in the VE, VF, UK and UF system. The raw water of the VF-system is chemically treated with sodium hypochlorite to prevent growth and accumulation of bacteria, mussels and other organic substances, as well as with ferrous sulfate, to protect copper-alloyed materials. The components in VE, VF, UK and UF system are subject to regular visual inspections through plant walkdowns. These walkdowns are implemented to detect leakages in an early stage. Based on these arguments it is concluded that crevice corrosion is adequately managed for all in-scope subcomponents and components [19] [20] [21].

3.6.9 Flow accelerated corrosion

Flow Accelerated Corrosion (FAC) is a degradation process for metallic materials that generally are corrosion resistance due to the formation of protective oxide layers. The protection offered by the oxide layer is reduced or eliminated by the velocity of the medium. FAC depends on water chemistry (in particular dissolved oxygen and pH), temperature, flow conditions, material composition, component geometry and exposure period. Carbon steel and low-alloy steel are generally susceptible to FAC. Carbon steel and low-alloy steel containing < 2 wt.-% Cr may be susceptible to FAC when subjected to flowing high-temperature water with a dissolved oxygen concentration < 50 ppb and pH < 9,5. It is generally accepted, that FAC mainly appears in the temperature range of 50 °C to 250°C [7]. In special cases, normally not valid for the in-scope components, FAC can also occur at lower temperatures. For conservatism, the AMR applied a lower limit of 40 °C.

KCBs primary circuit is resistant to FAC based on material selection, i.e., the application of austenitic stainless steel and nickel-base alloys for surfaces in contact with main coolant water. However FAC was identified as a relevant ageing mechanism for the low-alloy steel SG feedwater and main steam nozzles, as well as part of the secondary system components (see Table 3-7).

KCB implemented a FAC program. This program determines the susceptibility and extent of FAC in certain system components and subcomponents (e.g., piping, valve bodies, pipe elbows, and expanders) composed of carbon steel and low-alloy steel. The FAC program manages FAC through the:

- Prevention of FAC (e.g., controlling and monitoring of water chemistry and pH);
- Analysis to determine critical locations;
- Performance of baseline inspections to determine the extent of thinning at the critical locations;
- Performance of follow-up inspections to confirm thinning rate predictions;
- Repair or replacement of components as necessary, taking into account possible improvements in the geometric layout, material selection, or general operating conditions of the component in question.

The scope of the FAC program is determined by the outcome of the analysis of critical locations. System components that are identified as possibly susceptible to FAC are further investigated. Components of the following systems; RA, RB, RF, RG, RH, RK, RL, RM, RN, RT, RU, RY, and SD are included in the FAC program [43]. Plant walkdowns (e.g., operator plant walkdowns) are performed throughout the plant, and provide additional assurance that degradation or leakage of components due to FAC will be detected in time.

Steam generator

The SG feedwater (RL) and main steam (RA) nozzles are not part of the FAC sampling program, as they have previously, from 1975 on, been considered as low-risk components. In 1997 the feedwater and main steam line were replaced for LBB pipeline. Nevertheless it recommended in the SG AMR that these nozzles are again evaluated within the scope of a FAC screening analysis to determine wall thinning effects in terms of LTO [14].

Secondary systems

In the AMR it is concluded that FAC of secondary system components is adequately managed with the above listed activities [20].

Table 3-7 AMR results for the components identified as susceptible to FAC

(Sub)components	Recommendations
Steam generator	
Low-alloy steel feedwater and main steam nozzles including their respective safe ends.	Evaluate in FAC screening analysis
Secondary systems – RL, RA, RB, RT and RY systems	
Components made of cast iron, carbon steel and low-alloy steel in contact with treated water or steam for systems operated at a temperature range above 40 °C.	-

3.6.10 Denting

Denting is the mechanical deformation or constriction of SG tubes at a tube support plate connection. This ageing mechanism is caused by the buildup and growth of voluminous corrosion products in the annulus between the U-tube and tube support plate. Due to the design of KCBs SG tube supports denting is unlikely in this region. Although the design of the tube-to-tubesheet connection involved minimization of the gap size, denting of the U-tubes in the area at the top of the tubesheet cannot be excluded in the areas where hard sludge build-up can occur (see Table 3-8).

The water chemistry on primary and secondary sides of KCBs SGs is monitored and water chemistry measures are taken to minimize corrosion product transport into and accumulation on the secondary side of the SGs, as well as to assure the presence of reducing conditions. Additional conservation activities (wet or dry lay-up) during outage periods also serve to protect surfaces and limit oxygen ingress. As stated in section 3.6.2, the material condition of KCBs SG U-tubes is determined every 3 years. Recent inspections have shown no denting has been established in the area of concern. However in order to adequately manage denting for LTO, it is recommended in the SG AMR [14] to determine the level of deposit accumulation on top of the tubesheet (see also section 3.6.2).

Table 3-8 AMR results for the components identified as susceptible to denting

(Sub)components	Recommendations
Steam generator	
External surfaces of the U-tubes	Ensure that the presence and height of deposits that have accumulated on top of the tubesheet are measured and determine if sludge removal is needed

3.7 Management of ageing by corrosion with additional mechanical loading

3.7.1 Intergranular stress corrosion cracking

Intergranular stress corrosion cracking (IGSCC) is characterized by the progressive nucleation and propagation of cracks preferentially along the grain boundaries due to localized corrosion induced by mechanical stress. Typically, the appearance of fracture surfaces shows the surfaces of individual grains.

Intergranular stress corrosion cracking of carbon steel or low-alloy steel is not relevant for PWRs if operated under specified conditions. In general, IGSCC does not affect austenitic stainless steel components or subcomponents in the primary or secondary loop of a PWR during normal service conditions, as the dissolved oxygen content is below 10 ppb. However, research results show that IGSCC can occur in non-thermally sensitized microstructures in low electrochemical potential conditions in the event that severe cold working of a material has occurred [7].

For small piping (e.g., instrumentation piping, measurement lines, vent lines, drain lines) the relevant manufacturing documents (particularly welding or heat treatment plans) were not reviewed in the AMR. Given the fact that small piping is generally cold worked during the fabrication process, according to existing standards, the occurrence of IGSCC cannot be completely excluded. Table 3-9 lists all components, mostly small piping, for which IGSCC is identified as a relevant ageing mechanism.

In the event that leakage of small piping would occur, it can be identified through visual inspections during plant walkdowns (the external surfaces for all accessible components of the main coolant system are visually inspected during the leakage testing walkdowns (VT-2) following each outage prior to plant start-up, while the external surfaces of remaining systems are looked at during the regular plant walkdowns) and/or the leakage monitoring system. Visual inspection and leakage monitoring are defined as sufficient to manage IGSCC of small piping [11][13][14][17][19][20][21][22]. The AMR results of the remaining components are described below.

Steam generator

IGSCC of the alloy 800 U-tubes is described in section 3.7.5 “Outer diameter stress corrosion cracking (ODSCC)”, as that name is most frequently used in the industry to describe and handle the occurrence of this degradation mechanism [14].

Main coolant pump

Thermal sensitization of some welds of the upper partial inserts and for the tube elbows located in the HP cooling circuit (emergency seal water supply lines) and the HP and LP leakoff lines cannot be excluded, and therefore IGSCC is a relevant ageing mechanism. As KCB performs periodic liquid penetrant testing of a certain fraction of welded joints in piping YD001/002 Z004, Z005, Z006 every 10 years IGSCC is adequately managed [15].

Safety-related auxiliary systems

The heat exchanger internals for the heat exchangers TV061/TV062 B001 and TA000 B001/B002 are exposed to treated borated water and operating temperatures > 200 °C. Therefore, these internals may be susceptible to IGSCC due to thermal sensitization. As KCB measures the temperature in the TA system downstream of the recuperative heat exchanger (tube-side of TA000 B001/B002), failure of heat exchanger tubes can be identified through variations in the heat exchanger outlet temperature. With respect to the sample heat exchangers (TV061/TV062 B001), the sample lines TV061/062 are pressure tested annually and in addition failure of TV heat exchanger tubes can be identified through variations in the heat exchanger outlet temperature [21].

Reactor pressure vessel internals

IGSCC of cold worked stainless steel could occur at overall low ECPs. Therefore, IGSCC is relevant for the baffle-former bolts and barrel-former bolts made of cold worked austenitic stainless steel. These bolts are inspected at KCB by means of VT and UT. As these inspections were not recommended by the supplier of the plant in the past, they are performed as ad-hoc inspections. With these inspections IGSCC of the RPV internals is adequately managed [23].

Table 3-9 AMR results for the components identified as susceptible to IGSCC

(Sub)components	Recommendations
RPV	
Relevant pipe sections of heat affected zones or cold working areas in the austenitic stainless steel intermediate extraction line and vent line.	-
Pressurizer	
Austenitic stainless steel instrumentation piping, measurement lines, vent lines, drain lines, and RPV supply lines.	-
Steam generator	
Austenitic stainless steel drain piping	-
External surfaces of the U-tubes	See section 3.7.5
Main coolant pump	
Internal surfaces of welds and tube elbows in the HP cooling circuit (emergency seal water supply lines) and HP and LP leakoff lines.	-
Main coolant lines and pressurizer surge line	
Small gauge austenitic stainless steel piping	-
Nuclear safety systems Secondary systems Safety-related auxiliary systems HVAC systems	
Forged austenitic stainless steel small bore piping	-
Heat exchanger internals of TV061/TV062 B001, TA000 B001/B002	-
RPV internals	
Baffle-former and barrel-former bolts	-

3.7.2 Primary water stress corrosion cracking

Primary Water Stress Corrosion Cracking (PWSCC) is a type of intergranular stress corrosion cracking which may occur in nickel-base alloys subjected to high-temperature main coolant water. Precondition for the occurrence of PWSCC is the simultaneous presence of a susceptible material condition, high tensile stresses and a corrosive environment. The chemistry program is especially designed to mitigate PWSCC by keeping the hydrogen content within a specific range. In PWR primary water conditions, PWSCC may occur in alloy 82, 182 and 606 welding filler metals and alloy 600 [7]. Table 3-10 lists the in-scope components that were identified as susceptible to PWSCC. The AMR results of these components are described below.

Reactor pressure vessel

PWSCC is a relevant ageing mechanism for the core barrel guide blocks and associated welded joints between the internal cladding of the vessel flange on the upper cylindrical shell and the core barrel guide blocks (Figure 3-2). If any crack initiates in the core barrel guide blocks welds, it may affect the integrity of the internal cladding of the RPV. Therefore, it is necessary that the associated welds of the core barrel guide blocks are periodically inspected. KCB performed the last visual inspection (VT-3) in 2004. However, VT-3 only determines the overall mechanical and structural condition of the core barrel guide blocks and its supports and is therefore not considered adequate to detect defects in the inspected area.

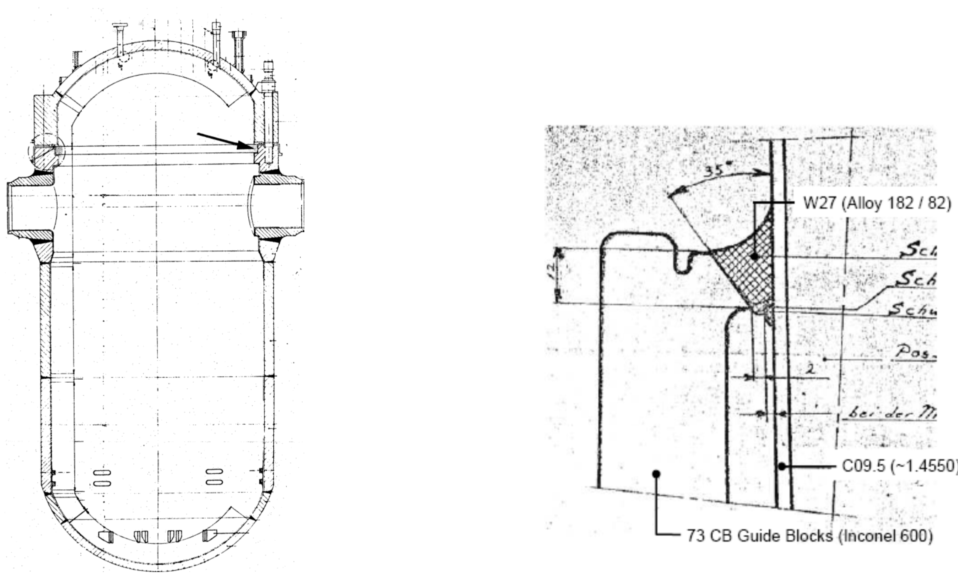

Figure 3-2 Left sketch RPV (arrow indicates core barrel guide blocks) and right sketch of core barrel guide block weld

In the AMR of the RPV it is recommended that KCB performs a VT-1 inspection of this location (in particular the weld joints) to determine the surface condition and to identify any defects or degradation effects, such as cracking, erosion, abrasion, corrosion or wear. Furthermore, if cracks are detected during ISI activities and found to be beyond acceptance criteria, the core barrel guide blocks and associated weld joints should undergo further investigation, to determine if replacement with more PWSCC resistant components or weld joints is required. With this additional activity PWSCC of the RPV is adequately managed [11].

Pressurizer

PWSCC is a relevant ageing mechanism for the C-torus manway seal (made of alloy 600 and welded with alloy 82 welding filler metal). If PWSCC of the C-torus seal were to occur, it could eventually lead to leakage of main coolant water in the sealing area, which would be detected by the VT-2 inspections before start-up and/or by the leakage monitoring system. The potential for boric acid corrosion of the fasteners and the manway cover due to leakage through the C-torus seal assembly is managed by the boric acid corrosion program. In addition an inspection of the manways is included in the ISI program. Therefore, it is concluded that PWSCC of the pressurizer is adequately managed [13].

Steam generator

PWSCC is a relevant ageing mechanism for the welded tube plugs (alloy 600) and the welds and cladding (inconel 82 and alloy 606) on the tubesheet. The welds between the tubesheet and channel head ring are subjected to regular ultrasonic testing. During these testing activities the condition of the cladding in this area is identified. As any crack initiation or crack growth in the weld joints on the tubesheet could affect the cladding of the Main Coolant Pressure Boundary (MCPB) it is necessary that the associated welds and cladding on the tubesheet are periodically inspected. Therefore, the SG AMR recommends additional VT-1 inspections for the welds and cladding on the tubesheet to determine the surface condition and to identify any defects or degradation effects. To detect any defects or degradation of the welded tube plugs, it is recommended in the SG AMR to perform a VT-1 inspection, when the primary side of the SG is open for inspection. With these additional activities PWSCC of the steam generator is adequately managed [14].

Reactor pressure vessel internals

PWSCC is a relevant ageing mechanism for the bolts of the in-core instrumentation tubing, the lower fuel assembly centering pins and the baffle-former bolts in the lowest row. PWSCC of the distance pieces of the schemel (alloy 600) is very unlikely due to the low stress level. The fuel assembly centering pins are annually inspected (YH110 / YH 206) and the baffle-former bolts are visually inspected and ultrasonic tested. In case PWSCC would occur it can be identified using the loose parts monitoring system. Therefore, it is concluded that PWSCC is adequately managed for the RPV internals [23].

Table 3-10 AMR results for the components identified as susceptible to PWSCC

(Sub)components	Recommendations
RPV	
Core barrel guide blocks and associated weld joints	PerformVT-1inspection
Pressurizer	
C-torus manway seal	-
Steam generator	
- Welded tube plugs; - Nickel-base alloy welds and cladding on the tubesheet.	PerformVT-1 inspection
RPV internals	
- Bolts of the in-core instrumentation tube; - Lower fuel assembly centering pins; - Baffle-former bolts in the lowest row.	-

3.7.3 Transgranular stress corrosion cracking

Transgranular stress corrosion cracking (TGSCC) is characterized by the progressive nucleation and propagation of cracks preferentially through grains. Nickel-base alloys are generally highly resistant to TGSCC. However austenitic stainless steel with a nickel content below approximately 15 wt.-% is susceptible to TGSCC when exposed to water containing critical amounts of chlorides and dissolved oxygen. A special case of TGSCC of austenitic stainless steel is atmospheric stress corrosion cracking. In this case, contamination by airborne chlorides at seawater sites can be caused at ambient temperatures.

TGSCC is mitigated in the primary circuit and secondary cooling systems at KCB as the presence of high chloride and oxygen concentrations is mitigated by implementation of the KCB water chemistry program. During normal operation, the concentration of dissolved oxygen in main coolant water is below 0,001 ppm and the chloride concentration is below 0,005 ppm. These values are far below the critical concentrations for TGSCC [7]. Additionally, the presence of chlorides (e.g., from gaskets or fitting lubricants), is nowadays managed at KCB through the use of materials that have a low chloride content. Contamination of lubricating oil with water, dissolved oxygen and other impurities is monitored through the quality analysis of the lubricating oil.

Besides these preventive activities, the external surfaces for all accessible components of the main coolant system are visually inspected during the leakage testing walkdowns (VT-2) following each outage prior to plant start-up and continuously monitored during operation by the leakage monitoring system. In this way the integrity of the pressure boundary is inspected and corrective actions are in place if adverse conditions are identified. The RPV internals are annually visually inspected and the external surfaces of remaining systems are looked at during the regular plant walkdowns. These walkdowns are performed to identify in an early stage degraded components during plant operation. With these activities in place it is concluded that TGSCC is adequately managed for all in-scope structures and components [10] [11] [13] [14] [15] [16] [17] [18][19] [20] [21] [23] [24] [25].

3.7.4 Irradiation assisted stress corrosion cracking

Irradiation Assisted Stress Corrosion Cracking (IASCC) refers to the intergranular cracking of materials exposed to ionizing radiation and fast-neutron irradiation. Annealed and irradiated austenitic stainless steel and nickel-base alloys become susceptible to IASCC when certain criteria (i.e., threshold fluence levels as a function of stress level, aggressive conditions, critical stress levels) are exceeded. The primary criterion for IASCC involves exceeding a critical threshold, i.e., the accumulated fast-neutron fluence. This threshold is 2×10^{21} n/cm^2 (E > 1 MeV) for PWRs [7]. As this threshold can be reached during LTO, IASCC is identified as a relevant ageing mechanism for the RPV internals at KCB [23].

Studies have shown that this threshold is already exceeded for the core baffle, baffle-former bolts and former plates along the core main axis (0°) at the axial position of the core midplane [44]. Another study shows the axial fluence distribution after 40 and 60 years of operation [45]. As this study calculated the axial fluence distribution starting from cycle 1 and the baffle-former bolts were replaced in 1988, it is recommended in the AMR to verify the fluence to be expected over the cycles 15 - 60. As an outcome of this verification, it should be established if and when IASCC could become a possible relevant ageing mechanism for the baffle-former bolts. Based on this outcome, the inspection plan should possibly be modified [23].

IASCC is mitigated by KCBs water chemistry program, which monitors the concentration of chemical species, including hydrogen and adjusts the chemistry properties if needed. However, proper water chemistry cannot fully suppress IASCC. To adequately manage the possible occurrence of IASCC (e.g., fracture) of in-scope RPV internal components and subcomponents, KCB has several activities in place. These activities include activities performed during inspections aimed at detecting other ageing mechanisms, as well as detection of loose parts in the primary circuit, using the loose parts monitoring (KÜS) system. In addition KCB performed several ad-hoc inspections, especially VT and UT inspections of the barrel-former and baffle-former bolts With these activities in place and the verification of the expected fluence over cycle 15 – 60, IASCC is adequately managed for the RPV internals [23].

Table 3-11 AMR results for the components identified as susceptible to IASCC

(Sub)components	Recommendations
RPV internals	
- Core baffle; - Core barrel; - Former plate; - Barrel-former bolts; - Upper and lower fuel assembly centering pins; - Lower core support; - Grid plate; - Bolts of in-core instrumentation tube.	
- Baffle-former bolts.	Verify the expected fluence over the cycles 15-60 to determine if and when IASCC could become a possible ageing mechanism for the replaced baffle-former bolts.

3.7.5 Outer diameter stress corrosion cracking

Outer diameter stress corrosion cracking (ODSCC) is a specific case of corrosion which is relevant at the external surface of the SG U-tubes. The fracture mode of ODSCC could be either transgranular or intergranular, but it is most often observed as intergranular [7].

To mitigate the occurrence of ODSCC, it is necessary that the steam generators maintain a high level of cleanliness. To minimize corrosion product transport and accumulation on the secondary side of the SGs water chemistry measures are taken. As stated in section 3.6.2, the material condition of KCBs SG U-tubes is determined every 3 years using Eddy current testing. In addition Eddy current array testing (X-probe) and ultrasonic testing on the U-tubes are applied. Eddy current testing can detect the presence and height of deposits that have accumulated on top of the tubesheet. It is recommended in the SG AMR to ensure that these measurements are part of the testing procedure. Based on the determination of expected deposit quantities KCB should determine the necessity for sludge removal from collection areas on the top of the tubesheet [14].

Table 3-12 AMR results for the components identified as susceptible to ODSCC

(Sub)components	Recommendations
Steam generator	
External surfaces of the U-tubes	Ensure that the presence and height of deposits that have accumulated on top of the tubesheet are measured and determine if sludge removal is needed.

3.7.6 Hydrogen induced stress corrosion cracking

Hydrogen Induced Stress Corrosion Cracking (HISCC) involves the interaction of absorbed hydrogen formed during corrosion of the metal lattice. To initiate this ageing mechanism, an aqueous electrolyte must be present. HISCC is limited to high-strength ferritic steel with a yield strength > 800 MPa and ultimate tensile strength > 900 MPa. HISCC of martensitic stainless steel has been identified to occur at lower operating temperatures. Other materials are generally not susceptible to HISCC [7]. Table 3-12 lists the structures and components that were identified to be susceptible to HISCC. The AMR results of these components are described below.

Steel containment

HISCC is a relevant ageing mechanism for the high tensile bolts made of carbon steel or low alloy steel in case of leakage or a humid atmosphere. To detect HISCC it is recommended to analyze areas of the steel containment structure where high strength bolting materials are applied. These places should be visually inspected and it is recommended that in representative areas the strength of high strength bolt connections is inspected by torqueing [18].

Nuclear safety systems

HISCC is a relevant ageing mechanism for external surfaces of in-scope martensitic stainless steel components and internal surfaces of in-scope martensitic stainless steel components exposed to closed cycle cooling water at low temperatures (valve bodies TF015-TF017 S007). As KCB performs regular plant walkdowns aimed at identifying leakages, HISCC of these components is adequately managed [19].

Secondary systems

HISCC is a relevant ageing mechanism for external surfaces of in-scope martensitic stainless steel components and internal surfaces of in-scope martensitic stainless steel components exposed to closed cycle cooling water and lubricating oil at low temperatures (valve bodies VG091-093S020 and RL023S041). As KCB performs regular plant walkdowns aimed at identifying leakages HISCC of these components is adequately managed [20].

Mechanical fasteners

HISCC is relevant for bolts and nuts with a strength class of >8.8 and 8 made of carbon steel, low-alloy steel and martensitic stainless steel in an aqueous environment. As leakages can be identified using KCBs leakage monitoring systems and during the regular plant walkdowns, HISCC of high strength fasteners is adequately managed. However it is recommended in the AMR to verify that all areas with in-scope fasteners are covered by these activities. In the case that corrosion traces on high strength fasteners are identified during the plant walk downs, it is recommended to exchange the affected fasteners. Where austenitic stainless steel of quality A2 and A3 is used and the presence of chlorides cannot be excluded it is recommended to replace the parts of concern by fasteners made of A4 or A5 steel quality [25].

Table 3-13 AMR results for the components identified as susceptible to HISCC

(Sub)components	Recommendation
Steel containment	
Carbon steel or low alloy steel high tensile bolting.	Perform a visual inspection and check of the strength of high tensile bolt connections.
Nuclear safety systems	
External surfaces of martensitic stainless steel components and internal surfaces of TF015-TF017 S007.	-
Secondary systems	
External surfaces of martensitic stainless steel components and internal surfaces of RL023S041, VG091/092/093S020.	-
Mechanical fasteners	
Carbon steel, low alloy or martensitic stainless steel high strength fasteners in an aqueous environment.	Verify that leakages of all in-scope fasteners will be noticed using the leakage

(Sub)components	Recommendation
	monitoring system and/or plant walk downs. Replace austenitic stainless steel fasteners of steel quality A2 and A3 by fasteners made of A4 or A5 in case the presence of chlorides cannot be excluded.

3.7.7 Strain induced corrosion cracking

Strain Induced Corrosion Cracking (SICC) may attack carbon steel and low-alloy steel which undergo plastic deformation at low positive strain rates. The result is the occurrence of very small (local or global) plastic deformation and transgranular cracking. Carbon steel and low-alloy steel are generally susceptible to SICC if ambient temperatures are higher than 100 °C, the strain rate is in the range of 10^{-3} – 10^{-7} s^{-1} in the plastic deformation region, and the oxygen content in the medium is higher than 30 and 80 ppb [7]. SICC is identified as a relevant ageing mechanism for the structures and components listed in Table 3-14. The AMR results of these components are described below.

Steam generator

Carbon steel and low-alloy steel secondary side components and subcomponents may be susceptible to SICC. The water chemistry is continuously monitored during normal operation and conservation activities (wet or dry lay-up) serve to protect SG component surfaces and limit oxygen ingress during outage periods. This, in combination with KCBs operating practices effectively prevents the occurance of SICC. Therefore it is concluded that SICC is adequately managed for SG (sub)components [14].

Nuclear safety systems

SICC is a relevant ageing mechanism for the carbon steel heat exchanger internals of TN090B003. SICC occurs due to a critical combination of material, strain, environmental temperature and oxygen content of the medium [7]. The conclusion is drawn in the nuclear safety systems AMR [19] that even if stress were to occur at the critical combination of strain rate and environmental temperature, the materials used and the low oxygen (<<30 ppb) in the medium due to the control by the KCB water chemistry program exclude SICC. Therefore, SICC of heat exchanger TN090B003 is adequately managed [19].

Secondary systems

SICC has been identified as a relevant ageing mechanism for cast iron, carbon steel and low-alloy steel secondary system components and subcomponents exposed to temperatures higher than 100 °C.. SICC occurs due to a critical combination of material, strain, environmental temperature and oxygen content of the medium [7]. The conclusion is drawn in the secondary systems AMR [20] that even if stress were to occur at the critical combination of strain rate and environmental temperature, the materials used and the low oxygen (<<30 ppb) in the medium due to the control by the KCB water chemistry program exclude SICC. In the case SICC would occur, KCB will identify the degraded components in an early stage, as they perform regular plant walkdowns (e.g., operator plant walkdowns (1x per shift) and multidisciplinary walkdowns). For some components additional visual inspections, ultrasonic testing or magnetic particle testing are performed. Therefore, SICC of these components is adequately managed [20].

Safety-related auxiliary systems

SICC has been identified as a relevant ageing mechanism for TN061/062S001. SICC occurs due to a critical combination of material, strain, environmental temperature and oxygen content of the medium [7]. The conclusion is drawn in the safety-related auxiliary systems AMR [21] that even if stress were to occur at the critical combination of strain rate and environmental temperature, the materials used and the low oxygen (<<30 ppb) in the medium due to the control by the KCB water chemistry program exclude SICC. In the case SICC would occur, KCB will identify the degraded components in an early stage, as they perform regular plant walkdowns (e.g., operator plant walkdowns (1x per shift) and multidisciplinary walkdowns). Therefore, SICC of these components is adequately managed [21].

Table 3-14 AMR results for the components identified as susceptible to SICC

(Sub)components	**Recommendations**
Steam generator	
Carbon steel and low-alloy steel secondary side components and subcomponents.	-
Nuclear safety systems	
The external surfaces of carbon steel heat exchanger	-

(Sub)components	Recommendations
internals (TN090B003).	
Secondary systems	
Cast iron, carbon steel and low-alloy steel secondary system components and subcomponents exposed to temperatures > 100 °C.	-
Safety-related auxiliary systems	
Carbon steel components and subcomponents exposed to temperatures > 100 °C (TN061/TN062 S001).	-

3.7.8 Corrosion fatigue

In Light Water Reactor (LWR) coolant environments an environmental effect on the fatigue life of low-alloy steel and austenitic stainless steel was shown by laboratory experiments. Thus far, there has been no documented occurrence of fatigue failures in an operating LWR plant where the cause of the failure can be attributed to a reduction in fatigue life due to LWR coolant environmental effects. However, since experimental evidence indicates that a reduction in fatigue life could occur in certain conditions, environmental effects should be considered during plant operation.

Effects of the LWR coolant environment on the fatigue resistance of a material are currently not explicitly addressed. According to a draft version of KTA 3201.2 one of the following measures shall be considered in the event that uncertainties exist with respect to environmental effects:

- Limitation of the Cumulative Usage Factor (CUF);
- Relevant laboratory testing;
- Appropriate measures regarding in-service inspection and operational monitoring (e.g., incorporation of the relevant component into a fatigue monitoring program).

According to this draft KTA, actions shall be taken if environmental effects cannot be excluded and the CUF reaches the following fixed threshold values:

- CUF = 0.2 for austenitic stainless steel;
- CUF = 0.4 for ferritic steel (carbon steel and low-alloy steel).

In the AMR it is recommended that KCB evaluates the CUF for all relevant components and subcomponents (see section 3.5) based on the latest KTA rules (currently in draft form) with respect to environmental effects, as well as to follow any future, international developments on this issue.

Note: The consideration of environmental effects is not necessary for components and subcomponents that are not in contact with coolant water (e.g., studs of the RPV vessel flange).

3.8 Management of other ageing mechanisms

3.8.1 Change in material properties

Thermal exposure of elastomers can degrade the material microstructure, adversely affecting the characteristics of the material [7]. For components of the EY system exposed to a temperature > 90 ^{0}C and treated water or lubricating oil (expansion bellows in EY033/EY053 and flex connections in EY014-EY054/EY053), thermal exposure cannot be ruled out. Therefore, change in material properties is a relevant ageing mechanism for these components.

As KCB performs regular visual inspections through plant walkdowns and more specific inspections within the diesel maintenance scope (according to MTU procedures), this ageing mechanism is adequately managed. Furthermore, the expansion bellows and flex connections in the EY system are replaced every 8 years during the annual maintenance activities by MTU. With these activities in place ageing by thermal exposure is adequately managed [19].

3.8.2 Blistering

Coatings do not form an impermeable barrier. Therefore moisture can penetrate the coating and contact the underlying metal or concrete surface in areas with low adhesion. During the application of a coating, the solvent could be "trapped" on the coating-substrate interface, creating blisters, if evaporating time was not considered for the previous layer prior to application of next layer. Surface contamination beneath the coating acts as a collection site for moisture that has permeated through the coating thickness. Soluble salts have an affinity for moisture and draw the moisture toward the deposits. Hard contamination can also cause damage to the coating.

Blistering typically leads to localized corrosion of the substrate beneath the pore or blister. Soluble salts, in particular, can create conditions conducive to corrosion attack of the steel by lowering the pH value [9]. Blistering has been identified as a relevant ageing mechanism for the components listed in Table 3-15.

Blistering of components in small diameter piping (DN < 100 mm) is adequately managed through the performance of regular plant walkdowns. These walkdowns are aimed at identifying degraded components during plant operation. The AMR results of remaining components listed in Table 3-15 are described below.

Nuclear safety systems

Blistering has been identified as a relevant ageing mechanism for the coating on concrete surfaces of TW012/022B001, the cast iron rubber-lined valve bodies in the VE system, as well as the cast iron rubber-lined piping, instrumentation tubing and valve bodies in the VF system. As stated above blistering of components in small diameter piping is adequately managed through the performance of regular plant walkdowns. Besides these walkdowns KCB performs additional visual inspections (VT-2) of the coating on the concrete surfaces of TW012/TW022B001. The internal surface of coated VF piping and valve bodies in building 03, 04 and 21 has been visually inspected on several locations using video cameras. Failure indications were found and were repaired as necessary. Because degradation was identified in these representative areas, KCB will broaden the inspection scope to determine the extent of condition for all similar areas. With these activities in place it is concluded that blistering is adequately managed [19].

Secondary systems

Blistering has been identified as a relevant ageing mechanism for the coating on concrete surfaces of several tanks as well as several cast iron rubber lined valves in the UK system. As stated above blistering of components in small diameter piping is adequately managed through the performance of regular plant walkdowns. Besides regular plant walkdowns KCB performs additional visual inspections (VT-2, VT-3, VT of the inside surface) of the coating on concrete surfaces of RS011/021B002, RZ001/002/003/004B001 and VG000B001. With these activities in place blistering is adequately managed [20].

Safety-related auxiliary systems

Blistering has been identified as a relevant ageing mechanism for several carbon steel rubber lined valves in the TR and UW system as well as several carbon steel rubber lined tanks. For these components visual inspections through plant walkdowns are sufficient to manage blistering adequately [21].

HVAC systems

Blistering has been identified as a relevant ageing mechanism for several carbon steel rubber lined valves in the UV system and several expansion bellows in the UW system. For these components visual inspections through plant walkdowns are sufficient to manage blistering adequately [22].

Table 3-15 AMR results for the components identified as susceptible to blistering

(Sub)components	**Recommendation**
Nuclear safety systems	
- Coating on concrete surfaces of TW012/022B001; - Cast iron rubber-lined valve bodies in the VE system; - Cast iron rubber-lined piping, instrumentation tubing and valve bodies in the VF system.	-
Secondary systems	
- Coating on concrete surfaces of RS011/021B002, RZ001-004B001 and VG000B001; - Cast iron rubber lined valve bodies in the UK system (UK010 S001-S005, UK011/UK012 S001/S002 and UK013S001).	-
Safety-related auxiliary systems	
- Carbon steel rubber lined valves in the TR and UW system; - Carbon steel rubber lined tanks (TR011-014B001, TR041/042B001).	-
HVAC systems	
- Carbon steel rubber lined valves in the UV system; - Glass fiber coated with silicone expansion bellows in the UW system.	-

3.8.3 Cracking and delamination

Cracking may penetrate the top layer of a coating, and may propagate into lower layers of a coating or substrate. Cracking can provide a direct path to the substrate for moisture and chemicals, negating the protective function of the coating in the proximity of the crack [8].

The components identified to be possibly susceptible for blistering are also susceptible to cracking and delamination (see Table 3-15). Cracking and delamination is adequately managed by the activities described in section 3.8.2.

3.8.4 Ageing of thermoplastic materials

Ageing of thermoplastic materials can basically be divided into two categories:

- Physical ageing processes;
- Chemical ageing processes.

Physical ageing processes cause a change in the morphology of the material e.g., change of the crystal structure. These processes do not affect the chemical structure of the molecular chains whereas chemical ageing processes affect the chemical structure of the molecular chains. Physical ageing processes are for example relaxation of residual stresses and orientations of the polymeric chains, migration and loss of softeners (especially PVC) and post crystallization. The most important influence factors for the ageing of PE, PP and PVC for the application at KCB are:

- Temperature;
- Concentration of oxygen;
- Medium;
- High-energy irradiation;
- Mechanical loads.

The main effect of ageing of thermoplastic materials that impairs the long-term usability is the reduction of the molecular weight resulting in the decrease of the initial mechanical properties of the materials. Ageing of thermoplastic has been identified as a relevant ageing mechanism for the components and subcomponents listed in Table 3-16. The AMR results of these components are described below.

Safety-related auxiliary systems

The polypropylene piping sections TR010Z004/005/006/007 are originally designed for maximum operating temperature of 70 ^{0}C and a maximum operating pressure of 2,9 bar. Due to a specification change in 1980 these operating conditions do no longer fall within the normal design conditions according to the new RE-L 3377 specification. In the AMR it is therefore recommended to evaluate the degree of material deterioration. Based on the results of this evaluation KCB should determine whether replacement of the piping is necessary [21].

HVAC systems

The PVC measurement vessel (TM003B001) and the valve TM003S002 are susceptible to ageing of thermoplastics. Visual inspections through plant walkdowns are sufficient to adequately manage this ageing mechanism [22].

Table 3-16 AMR results for components identified as susceptible to ageing of thermoplastic materials

(Sub)components	Recommendation
Safety-related auxiliary systems	
Polypropylene piping TR010Z004/005/006/007	Evaluate the degree of material degradation and determine if replacement is necessary.
HVAC systems	
TM003B001 and TM003S002	-

3.8.5 Concrete shrinkage

In order to convey forces via the lower support (construction) of the PZR and MCP into the concrete structure, specified pretension of the tension studs is required. The loss of concrete moisture causes the concrete to shrink over the years, which could reduce the pretension of the tension studs. It is recommended in the AMR to check the pretension of the pressurizer and MCP lower support (construction) tension studs every 8 years. If the pretension of the tension studs is below the minimum specified value, the pretension should be adjusted as specified in the design specification or design calculation [10].

3.9 Other recommendations

3.9.1 Inspection of primary supports and their respective welds

ASME Section XI Division 1 Table IWB-2500-1 recommends a surface/volumetric examination of the support welds for class 1 components during each ISI interval. The remaining area of each in-scope support should, according to ASME section XI Section IWF, be visually examined (VT-3). For several mechanical A components these ASME requirements resulted in additional AMR recommendations. These recommendations are described below.

Reactor pressure vessel

The RPV support blocks were last inspected (UT-V) in 1994 and 1999. Although they are normally inspected every 10 years, they are currently not planned for inspection according to the ISI plan. It is recommended in the RPV AMR [11] to follow ASME XI, meaning performing a surface/volumetric examination of the RPV support welds during each ISI inspection interval. The AMR concluded that a visual inspection of the support blocks was not necessary (see also section 3.9.2) [10].

Pressurizer

A review of the information in the ISI and ISH database revealed that the welds which attach to the supports of the pressurizer, or the supports themselves have not been inspected after initial component installation. It is recommended to follow ASME XI, which means perform a surface or volumetric examination of the support welds for class 1 components, and perform VT-3 of the supports during each ISI interval[4] [13] [10]. In addition it is recommended to monitor that the actual PZR support assembly relative to the drawings assembly [10].

Steam generator

The SG02 support welds were last inspected volumetrically (UT) in 2006. The next inspection is scheduled for 2016. Therefore, it is concluded that the area of the support weld is adequately managed. It is recommended to follow ASME XI, which means in addition a VT-3 of the supports should be performed during each ISI interval[4] [14] [10].

Main coolant pump

The MCP casing support welds were last inspected in 1995 (MCP 1) and 1998 (MCP 2). At the moment no further ISI activities are planned for the casing support welds. ASME Section XI recommends surface/volumetric examination of the MCP support welds and a visual examination (VT-3) of the MCP supports during each ISI interval. It is recommended in the AMR to follow ASME XI[4] [15] [10]. In addition it is recommended to monitor that the actual MCP support assembly relative to the drawings assembly [10]

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[4] Note: After each refueling outage, prior to startup KCBs inspection department performs a plant walk down which included VT-2 and VT-3 inspection of accessible parts of the primary components.

3.9.2 Check of primary support clearances

Acceleration of the primary components in horizontal and vertical direction and thus induced forces are limited by means of the specified clearances of the whip restraints. If the clearances are not sufficient the primary components could be blocked due to thermal expansion, which could result in additional stresses for the primary components and eventually damage. Insufficient clearance in a German power plant resulted for example in bearing damage of the MCP motor (this incident did not affect nuclear safety). In the AMR it is therefore recommended to check the clearance of the primary component support guides and whip restraints during cold and operating conditions, to assure that the clearances of the primary component support guides and whip restraints are still within specified tolerances. Checking the clearances of the RPV supports is not recommended. There are no mechanisms that would induce restriction of the clearances, since the RPV is designed to be a fixed point in the center of the plant, exposed to only symmetrical forces coming from both ends during normal operation. Therefore, the forces offset each other and no net force acts upon the RPV. In addition inspection of the clearances is extremely difficult due to the inaccessibility of the supports and the high dose rate. Former inspections showed that the support of the RPV can safely be excluded from further monitoring [10].

3.9.3 Underclad cracking steam generator

Underclad cracks are actually cracks below the cladding. Underclad cracking was observed in the tubesheet and channel head ring during the manufacturing process. Initial reference records showed that the size of these cracks were well below the values regarded as critical and therefore acceptable. In addition the SG underclad cracks were inspected multiple times in the seventies, showing no measureable changes in the existing underclad crack size. As these defects were detected prior to commissioning and given that no crack growth was noted during the inspections in the seventies, it can be concluded that these underclad cracks originate from the manufacturing process. Nevertheless it is advised in the AMR to confirm that no crack growth will occur during normal operation. For the RPV it is shown that negligible crack growth is likely to occur during LTO. As it is possible that the loading conditions and crack/component configuration for the SG underclad cracks may be bounded by those of the RPV underclad cracks, it is recommended in the AMR to perform an evaluation to determine if the evaluations for the RPV underclad cracks can be used to conclude that SG underclad cracks will also undergo negligible crack growth during LTO [14].

3.9.4 Inspection of the CRDM pressure housings

In the CRDM pressure housings AMR it is recommended to extent the ISI activities of the CRDM pressure housings to detect any signs of corrosive attack and to determine the current status of the housings. This extension should include inspections of the internal surface in critical areas, as well as inspections of the welds by means of non-destructive examination. The extension should be based on current KTA 3201.4 or ASME XI requirements, both requiring volumetric examination of 10% of the pressure housing welds within a period of 10 years [16].

3.9.5 Inspection of the weld in MCP HP cooling circuit

ASME Section XI Division 1 IWB-2420 and Table IWB-2500-1 recommends the inspection of certain welds in the HP cooling circuit. All circumferential piping welds that have a diameter smaller than DN 100 should be subject to surface inspections on:

- all terminal ends and joints where each pipe or branch run is connected to vessels
- all terminal ends and joints where each pipe or branch run is connected to other components where the stress levels exceed either a CUF of 0,4 or a certain stress intensity range.

Additional piping welds should also be selected for examination so that the total number of circumferential butt welds selected for examination equals 25% of the circumferential butt welds in the HP cooling circuit for the MCPs. As KCB has chosen ASME for to guide their ISI strategy, it is recommended in the MCP AMR to review the current inspection plan for the MCPs [15].

3.10 General recommendations

3.10.1 In-service inspection plan

The ageing management review of mechanical A components showed that the inspections of some components and subcomponents still needed to be determined. It is recommended that KCB ensures that the correct ISI interval is in place for these components and subcomponents.

3.10.2 “Instandhouding” database

According to the KCB maintenance strategy [30] all safety category 1 to 4 components and subcomponents should be part of preventive maintenance. However the AMR revealed that some components and subcomponents, for example some safety class 1 and 2 main coolant line valve bodies and orifices, were not listed in the instandhouding (ISH) database as covered by preventive maintenance. To ensure that preventive maintenance, such as ISI or time-based maintenance (GAO), is performed on the components and subcomponents in compliance with the KCB maintenance strategy, it is recommended to update the information in the ISH database. The information should be corrected wherever needed based on the findings in the AMR.

3.10.3 ISI plan versus ISH database

In the AMR of the mechanical A components the ISI plan and the ISH database, which covers all inspections, monitoring and measurements concerning ISI, surveillance and maintenance, revealed that there are inconsistencies between the two databases. For example, some visual tests are scheduled in the ISH database, but not in the ISI plan. It is therefore recommended that KCB updates the ISI plan and ISH database regularly or merges the ISI plan with the ISH database to eliminate the possibility of errors and guarantee the examination and testing coverage in terms of ISI.

3.10.4 System leakage testing walkdowns

After each refueling outage, prior to start-up, all accessible components are visually inspected (VT-2) to detect leakages. However the plant walkdown procedures for these VT-2 inspections only generally list the inspection areas and actions that need to be completed. To clarify which areas, rooms or components need to be inspected during the system leakage testing walkdowns prior to start-up, it is recommended in the AMR to revise the plant walkdown procedures.

4 Results electrical and I&C AMR

This chapter describes of the results of the electrical and I&C AMR [27]. As stated in [6] the electrical and I&C AMR has been performed on a commodity level, distinguishing the following commodities:

- Cables;
- Wires;
- Electrical connections;
- Electrical containment penetrations;
- Cable penetrations (floors, walls);
- Cabinets, racks;
- Cable trays.

For each commodity group the applied materials that are susceptible to ageing due to the stressors long-term temperature, cumulative radiation dose, moisture and/or electrical field, as identified in the AMR, are summarized in this chapter. Thereafter, the ageing management activities that serve to manage ageing during LTO are evaluated. In the case the AMR identified that the existing ageing management activity is not adequate or does not exist at KCB recommendations are provided in this chapter as well. The implementation of these recommendations is described in chapter 6.

Cables

Multiple materials are used as insulation and/or jacket material for low voltage and medium voltage cables. The conductor material is either copper (Cu), nickel (Ni) or a nickel-chromium alloy. The applied insulation materials may be susceptible to ageing if these materials are subjected to thermal and/or radiation stressors. The long-term temperature, as well as the cumulative radiation dose above which the applied insulation and jacket materials are susceptible to ageing are listed in Table 4-1. The conductor materials are not susceptible to ageing caused by thermal and radiation stressors [8]. In addition cable materials PT, TPC, PUR, PI, CU and materials PE, PVC, XLPE applied for medium voltage cables could be susceptible to ageing if these materials are exposed to a moisture environment [8].

Table 4-1 Overview of the insulation and/or jacket materials of cables

Material	Maximum long-term temperature	Maximum cumulative dose
Natural rubber (NR)	60 ^{0}C	30 kGy (conservative assumption)
Polyvinylchloride (PVC)	65 ^{0}C	200 kGy
Polyethylene (PE)	60 ^{0}C	1200 kGy
Cross-linked polyethylene (XLPE)	90 ^{0}C	1000 kGy
Silicone rubber (SIR)	200 ^{0}C	30 kGy
Polyimide (PI)	260 ^{0}C	2000 kGy
Chloroprene rubber (CR)	100 ^{0}C	20 kGy (conservative assumption)
Thermoplastic copolyester elastomer (TPC)	120 ^{0}C	100 kGy (conservative assumption)
Ethylene-vinylacetate-copolymer (EVA)	120 ^{0}C	50 kGy (conservative assumption)
Fluorinated ethylene propylene copolymer (FEP)	205 ^{0}C	0,5 kGy
Polypropylene (PP)	100 ^{0}C	50 kGy
Polyester (PT)	130 ^{0}C	100 kGy
Polytetrafluoroethylene (PTFE)	250 ^{0}C	0,5 kGy
Polyurathane (PUR)	80 ^{0}C	20 kGy
Ethylene-propylene-diene-monomer (EPDM)	120 ^{0}C	500 kGy

Cables that were taken into account in other procedures (i.e. the AUREST database to LOCA required components [48] and the 6kV cable replacement program [49]) were not considered AMR.

Using KCBs cable database several PVC/PVC and PE/PVC cables were identified as susceptible to ageing as the maximum allowable long-term temperature was lower than the comparison temperature[5]. In

[5] In the AMR 6 temperature and 3 radiation areas have been distinguished. The temperatures respectively radiation levels in these areas are conservatively chosen using measured or building design values. The comparison temperature and cumulative dose rate of a commodity material are defined by the

addition several PVC/PVC and SIR/SIR cables were identified as susceptible to ageing caused by radiation [27].

Thermal and/or radiation induced ageing of PVC/PVC and/or PE/PVC cables is characterized by the formation of small, elusive molecules that leave the polymer, causing the polymer to degrade with time. SIR/SIR become more brittle and less elastic due to intense cross-linking. The higher the temperature and/or radiation dose rate, the faster these reactions develop, resulting in darkening of the cable insulation and insulation embrittlement (and finally cracking) [8]. Insulation embrittlement could eventually result in the loss of the insulation function.

KCB inspects the cables near an active component if maintenance on these active components is performed. In addition cables exposed to high level temperatures or dose rates (hot spots) are included in a replacement program. Within this program the ageing condition of the removed cables is examined. This information is used to improve the ageing management activities in place for cables [50]. The replacement program concerns especially cables within the containment and steam relief valve rooms. Nevertheless it is recommended in the AMR to extent the current visual inspections to other cables in the corresponding function chain of an active component, focusing on the cables in high level temperatures and dose rates (hot spots). A five year interval of these inspections is sufficient. It is also suggested to implement an additional ageing management program for the PVC/PVC, PE/PVC and SIR/SIR cables that were identified as susceptible to ageing caused by thermal and/or radiation stressors and cables that could be susceptible to ageing in a moisture environment (e.g., medium voltage PVC cables) [27].

Wires

Polyvinylchloride (PVC), Polyethylene (PE) and Ethylene-tetrafluoroethylene-copolymer (ETFE) are used as the insulation materials and copper as the conductor material of wires. PE and PVC are susceptible to ageing (discoloration and embrittlement can occur) if these materials are subjected to long-term temperatures above 60 respectively 65 ^{0}C [8]. In addition PVC is susceptible to ageing if exposed to a cumulative radiation dose of more than 200 kGy [8]. As wires are only installed in the spreader rooms and in I&C cabinets they are not exposed to such temperatures and cumulative radiation doses. Therefore, based on these stressors, significant degradation of wires is not expected [27]. However insulation embrittlement, hardening, plasticizer leakage and discoloration has been observed in the past. KCB

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temperature and radiation area (for the commodities cables, connectors and terminals and electrical penetrations, self-heating is taken into account by adding 6,5 ^{0}C to the area temperature).

visually inspects the spreader wires every year. In addition ageing accelerated elongation tests of these wires are performed every five years. As no specific activities for the wires in the I&C cabinets are in place, it is recommended in the AMR to develop an ageing management activity for these wires [27].

Electrical connections

The electrical connections at KCB could have Epoxy-resin (EP), Fluorocarbon-rubber (FKM), Polyamide (PA), Silcon rubber(SIR) or Styrene-ethylene-butadiene-copolymer (SEBS) as insulation material. These materials are susceptible to ageing if exposed to large amounts of radiation (material design cumulative radiation dose EP and SEBS >100 kGy, SIR > 30 kGy, PA >20 kGy and FKM > 0,5 kGy) [8]. Ageing of these insulation materials results in darkening of the insulation and insulation embrittlement. The higher the radiation dose rate, the faster these insulation materials age [8]. Insulation embrittlement could eventually result in the loss of the insulation function.

At the moment it cannot be excluded that these insulation materials are installed in such high level radiation areas (e.g. inside the containment). Therefore, it is recommended in the AMR to determine all electrical connections inside the containment and investigate whether these materials are installed in such environments. If these materials are indeed installed in such environments an evaluation regarding ageing and experiences is recommended [27].

As stated in above visual inspections are carried out in the framework of maintenance of active components. These concern connectors of the active components or near to the active components. In the AMR it is recommended to extend these visual inspections also to the other connectors and in particular terminals of the corresponding function chain. A five-yearly interval of these inspections could be sufficient.

Electrical containment penetrations

Multiple materials are used as auxiliary, conductor or insulation material for the electrical containment penetrations at KCB. The auxiliary materials include FKM and SIR, which are susceptible to ageing if they are exposed to a cumulative radiation dose of more than 0,5 kGy respectively 30 kGy [8]. Radiation induced ageing of FKM is characterized by C-C-chain-scission and breaking of C-F-bindings. Small elusive molecules are formed that leave the polymer, causing the polymer to degrade with time. SIR under irradiation will become more brittle and less elastic due to intense cross-linking. The higher the radiation dose rate, the faster these reactions develop, resulting in darkening and embrittlement (and finally cracking) [8]. Therefore, it is recommended in the AMR to investigate if the materials FKM and SIR are installed on the containment side of the electrical penetrations. If these materials are applied on the containment side an evaluation regarding ageing and experiences is recommended [27].

The materials Cu, Cu-alloy (plating) and Ni-Fe-alloy, used as a conductor material for electrical containment penetrations can be susceptible to ageing processes, such as corrosion, when they are installed in a moist environment. Therefore, it is suggested in the AMR to implement an additional ageing management program for Cu, Cu-alloy and Ni-Fe-alloy electrical penetrations in a moist environment [27].

At the moment KCB performs regularly Helium-leakage tests regarding electrical containment penetrations. In the AMR it is recommended to add a visual inspection of the electrical insulation to these leakages tests [27].

Cable penetrations (floors, walls)

The in-scope cable penetrations include the pressed rubber penetrations (MCT Brattberg), the turbine building cable penetrations and the fire resistant barriers. EPDM, mineral wool, glass wool and refractory cement are used as sealing material, while iron/steel (Fe) and aluminum (Al) are used as the frame material for these cable penetrations. Only the material EPDM could be susceptible to ageing (discoloration and embrittlement) caused by thermal or radiation stressors. However the temperatures and radiation levels of the areas were cable penetrations are installed, are sufficiently low to exclude ageing of EPDM. Iron/steel can be susceptible to ageing processes, such as corrosion, if installed in a moisture environment. Corrosion will usually not affect the electrical integrity of cable penetrations. Nevertheless it is suggested in the AMR to visually inspect the cable penetrations every five years, assuring that for example damages or corrosion are identified in an early stage [27].

Cabinets and racks

Cabinets and racks are made of iron/steel and coated/plated with zinc. Ageing of zinc plating/coating due to corrosion or irradiation is not relevant for in-scope cabinets and racks. In case the zinc plating/coating is damaged, the iron/steel in a moisture environment is susceptible to ageing processes, such as corrosion [8]. To assure that damage of the zinc plating/coating or corrosion is identified in an early stage, it is recommended in the AMR to visually inspect the cabinets and racks every five years [27].

Cable trays

Cable trays are made of iron/steel and coated/plated with zinc. Ageing of zinc plating/coating due to corrosion or irradiation is not relevant for in-scope cable trays. In case the zinc plating/coating is damaged, the iron/steel in a moisture environment is susceptible to ageing processes, such as corrosion [8]. To assure that damage of the zinc plating/coating or corrosion is identified in an early stage it is recommended in the AMR to visually inspect the cable trays every five years [27]. As KCB operating experience showed that it is not clear at the moment if the mechanical load of the cable trays is still under the design limits, it is recommended to check the mechanical loads of the cable trays in addition [27].

General recommendations

KCB technicians usually inspect cables and connectors visually during their activities. However in some inspection documents these visual inspections are not explicitly mentioned. In the AMR it is therefore recommended to incorporate visual inspections in all relevant inspection documentation [27].

5 Results of the AMR for structural components

This chapter describes the results of the AMR for structural components [26]. As stated in [6] the AMR for structural components has been performed on all earthquake class I and IIA buildings and their structural SSCs, as well as several non specific building structures.

In the scoping process [2] the following buildings were identified as relevant for LTO:

- Steel containment (01) except the inner primary steel containment, which is handled as a mechanical A SC [18];
- Reactor building annulus (02);
- Auxiliary reactor building (03), including the secondary pressure relief station;
- Turbine building (04);
- Switchgear building (05);
- Ventilation chimney (13);
- Emergency diesel generator buildings (10, 72 and 33);
- Building for auxiliary water supply systems and emergency control building (33, 35);
- Cooling water inlet building 21 and cooling water outlet building 23.

The structural components identified during the scoping [2] and screening process [3], which are not related to a specific building are grouped as follows during this AMR:

- Ultimate heat sink system (VE);
  The structural, important to safety defined parts of the VE-system are the concrete structure including all embedded parts and lifting connections and its support. The VE-pumps and the water intake, as well as the feed water lines are part of the mechanical AMR.
- Piping and cable ducts;
- Crane tracks and lifting equipment (steel components, e.g., passive long-lived components used for moving heavy loads);
- Fire barriers.

For each building or commodity the applied materials that are susceptible a certain ageing mechanism given the environmental condition are identified in the AMR. Thereafter the ageing management activities that serve to manage ageing during LTO are described and evaluated. In the case the AMR identified that the existing ageing management activity is not adequate or does not exist at KCB recommendations are provided in this chapter. The implementation of these recommendations is described in chapter 6.

5.1.1 Relevant ageing mechanisms for in-scope buildings and structural SCs

The AMR identified for each building or commodity the applied materials that are susceptible to a certain ageing mechanism given the environmental conditions. As the list of relevant ageing mechanisms is rather extensive this section only provides some examples.

The roof of the primary containment, reactor building annulus and reactor auxiliary building is made of concrete and exposed to outdoor air (and in case of rain to raw water). The ageing mechanisms identified as relevant for this roof in the AMR include amongst others:

- freeze-thaw;
- corrosion for embedded steel and steel reinforcement;
- reaction with aggregates;
- fatigue.

The secondary pressure relief station, which is located on top of the reactor auxiliary building, is sheltered by a steel frame work with steel sheeting and bituminous roofing materials. The steel sheeting is made of galvanized steel and exposed to outdoor air. The ageing mechanisms identified as relevant for this sheeting are:

- general corrosion;
- crevice corrosion.

For a complete overview of all relevant ageing mechanisms see Chapter 5 of [26].

Note: In the catalog of ageing mechanisms [9], it is stated that a high neutron fluence in the presence of concrete structures may cause thermal warming of the concrete, growth of aggregates, and decomposition of inertial free water that could lead to degradation of the concrete matrix. Such a high neutron fluence could only exist near the biological shield. However in the AMR for structural components [26] irradiation was not identified as a relevant ageing mechanism for the concrete of the biological shield. The irradiation calculation of the RPV [23] shows that KCBs biological shield is not exposed a level of radiation were degradation of concrete is expected to occur during LTO.

5.1.2 Identified existing KCB ageing management activities

As part of KCBs preventive maintenance program KCB developed a civil inspection plan to implement ageing management between 2003 and 2008. This inspection plan is built up as a lattice, showing the main inspection programs at the top. The inspection plan contains in total 156 inspections (described in "werkomschrijvingen"; WNCs), that are set by external experts and consider operating experience at other plants, as well as the safety requirements for relevant components. In the AMR a complete overview of all relevant inspections is provided.

After an inspection is performed the condition of the relevant SSCs is reported in an inspection record. The responsible engineer evaluates the results and launches further investigations or measures if needed. External specialists and specialists within the KCB civil department consult each other to determine the correct measure.

5.1.3 Evaluation of ageing management

Based on the identified relevant ageing mechanisms for each building or commodity and existing KCB inspections the AMR concluded that ageing of KCB buildings and structural SCs is adequately managed. Recommendations for a pro-active inspection manner, are currently being implemented at KCB. The fulfillment of these recommendations at plant practices and/or policies align KCB with accepted nuclear industry practices (e.g., IAEA standards and guidelines) and demonstrate that the effects of ageing on SSCs important to safety will be adequately managed so that the intended function(s) will be maintained consistent with the KCB licensing basis during LTO.

6 Implementation of AMR recommendations

In chapters 0-5 the results of the mechanical AMR, the electrical AMR respectively the results of the AMR for structural components have been described. The mechanical, as well as the electrical AMR resulted in several recommendations regarding the specific areas in which KCB plant practices and policies should be augmented to align KCB with accepted nuclear industry practices (e.g., IAEA standards and guidelines) and to demonstrate that the effects of ageing on in-scope components and subcomponents will be adequately managed (i.e., the intended function(s) will remain consistent with the NPP licensing basis during LTO). The AMR concluded that the structural SCs are adequately managed with the existing ageing management activities.

Table B-1 in Appendix A summarizes all mechanical AMR recommendations. This table first lists the general recommendations, regarding for example the ISI plan and ISH database. Thereafter the recommendations for each individual SC (starting with the RPV) are listed. The SC specific recommendations are summarized in the third column. The first and second column provide the (sub)component(s) and, if applicable, the ageing mechanism(s) for which the recommendation has been made. In general KCB intends to implement all recommendations either by starting or finalizing an evaluation/study, or by planning, preparing and performing additional inspections. The implementation of each recommendation is described in the last column of Table B-1 in Appendix A.

The electrical AMR recommendations are summarized in Table B-2 of Appendix A. This table first lists the general recommendations. Thereafter the recommendations for each commodity group are listed. In general KCB intends to implement all recommendations by either revising existing procedures or by evaluating the necessity of a recommendation. Additional temperature/radiation measurements are for example planned in the next outage to verify that certain cables are susceptible to ageing. The implementation of each recommendation is described in the last column of Table B-2 in Appendix A.

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Appendix A Implementation of AMR recommendations

Table A-1 Mechanical AMR

<table>
<tr><th>SCs</th><th>Relevant ageing mechanism</th><th>Recommendation</th><th>Implementation</th></tr>
<tr><td colspan="4">General recommendations</td></tr>
<tr><td colspan="3">Ensure that the correct ISI interval is in place for mechanical A components and subcomponents (see § 3.10.1).</td><td>For components assigned as IAEA safety class 1 through 3, ISI activities are performed at the inspection intervals according to ASME Code Section XI Division 1 and Stoomwezen. KCB prepared a schedule for all inspection activities of the 5th inspection interval: January 2010 - December 2019 (10 years) [46]. Agreement with the Dutch regulator on the this inspection schedule has a high priority.</td></tr>
<tr><td colspan="3">Update the information in the ISH database and assure that IAEA safety class 1 to 4 components are part of preventive maintenance program (see § 3.10.2)</td><td>KCBs preventive maintenance strategy is one of the strategies, that assures that the condition of plant SSCs remains adequate. The other strategies include the in-service inspection, as well as the surveillance strategy. The programs resulting from these strategies each cover specific aspects. In order to prevent unnecessary duplications between the programs, the activities in these programs are aligned. In general safety class 1-4 (sub)components are part of preventive maintenance and/or periodic testing. KCB will evaluate whether all safety class 1-4 (sub)components, according to the ISH database, are managed in such manner. When needed the ISH database will be supplemented.
The ISH database also forms part of a larger project called</td></tr>
</table>

SCs	Relevant ageing mechanism	Recommendation	Implementation
			"iPOWER". In this project, KCBs existing maintenance and operating system (AS400) will be replaced with a new, state-of-the-art integrated data management system, using Ventyx software. This will result in improved traceability of relevant information.
Update the ISI plan and ISH database regularly or merge the ISI plan with the ISH database (see §3.10.3).			The ISI plan will be merged with the ISH database and integrated in the Ventyx system.
Revise the VT-2 plant walkdown procedure to clarify which areas, rooms or components are incorporated in system leakage testing walkdowns performed prior to start-up at the end of each outage (see § 3.10.4).			The existing procedure N12-22-VT4 [51] describes the visual inspections performed after each outage, prior to start-up. These visual inspections are aimed at leakage detection and changes in the general mechanical and structural condition of components and their supports. The main components and areas that should be considered during the inspection are listed in this procedure. This procedure was last updated in July 2011.
Reactor pressure vessel			
Relevant RPV subcomponents	(Corrosion) fatigue	Revalidate the existing fatigue analysis (§3.5) and consider environmental effects (§3.7.8).	The RPV subcomponents are considered in the TLAA fatigue project [52].

SCs	Relevant ageing mechanism	Recommendation	Implementation
Core barrel guide blocks and associated weld joints	PWSCC	Perform VT-1 inspection to determine the surface condition and identify any defects or degradation effects (§3.7.2).	KCB will add a VT-1 inspection of the core barrel guide blocks and their associated weld joints to the RPV inspections. The next inspection of the RPV is scheduled in the outage of 2015[6]. During refueling outages, in which the core barrel is not removed, KCB will inspect the visible part of the weld joints. The 2012 outage already provided an opportunity to inspect a large part of the welds. A report is currently being prepared by the AREVA inspectors.
RPV support blocks welds	-	Perform a surface/volumetric examination of the RPV support block welds during each ISI inspection interval (§3.9.1).	KCB normally examines the RPV support blocks using UT-V inspections. The RPV support blocks were last inspected in 1994 and 1999. Future inspections are currently not included in the ISI plan. KCB will ensure that the support blocks are UT-inspected during the inspections of the RPV, by adding this inspection to the ISI plan. The next inspection of the RPV is scheduled in the outage of 2015[6], when the core barrel will be out of the vessel.
Pressurizer			
Martensitic stainless steel nuts and washers in the pressurizer vent line/SEBIM valves	Thermal ageing	Assess possible ageing and consider replacement of these fasteners (§3.1).	KCB will assess the possible ageing of the martensitic stainless steel nuts and washers in the pressurizer vent line/SEBIM valves. Any modification of the design would only be contemplated in cooperation with the OEM (SEBIM) in order not to compromise the design qualification.

[6] Note: RPV inspections can only be performed during an extended outage (no reshuffle outage); after removal of the core barrel.

SCs	Relevant ageing mechanism	Recommendation	Implementation
PZR surge line nozzle at the PZR	Thermal fatigue	Re-evaluate the current loading configurations during normal operation (§3.5).	In 2010 the fatigue monitoring system (FAMOS) instrumentation was installed and implemented at KCB. The FAMOS system monitors thermal loading, in several fatigue relevant locations on the primary circuit and several points on the main feed water system piping. It is expected that KCB has gathered sufficient measurement information by the end of 2015 to calculate representative thermal loads on the surge line. In this process KCB will assess if there is any need to revise the existing operating procedures.
- PZR spray line - PZR spray line nozzles - Surge line nozzle at the PZR	(Corrosion) fatigue	Revalidate the existing fatigue analysis, validate assumptions using FAMOS (§3.5) and consider environmental effects (§3.7.8).	The pressurizer surge line, pressurizer spray line nozzles, as well as the surge line nozzle at the pressurizer are considered in the TLAA fatigue project [52].
Tension studs of the lower support construction	Concrete shrinkage	Check the pretension of tension studs every 8 years (§3.8.5).	During the refueling outage of 2012 KCB planned to check the pretension of the tension studs of the lower PZR support construction. However the insulation made these studs inaccessible. Therefore it was decided to postpone the investigation till the refueling outage of 2013 [53].

SCs	Relevant ageing mechanism	Recommendation	Implementation
PZR supports and welds that attach the supports to the PZR	-	Perform a surface/volumetric examination of the PZR support welds and a VT-3 inspection of the PZR supports during each ISI inspection interval (§3.9.1). Check that the PZR support assembly is in accordance with the assembly drawings (§3.9.1).	After each refueling outage, prior to start-up, the accessible components, including the pressurizer supports are visually inspected (VT-3) [49]. In addition a visual check of the condition of the PZR support was scheduled in the refueling outage of 2012. No indications were found. As the lower support was inaccessible, due to the insulation, this part will be investigated during the refueling outage of 2013 [53]. To ensure that any unexpected ageing mechanisms do not go unnoticed KCB will add a surface/volumetric examination of the connecting welds (PZR-supports) to the ISI program.
Pressurizer supports and whip restraints	-	Check the clearance of upper GAU support, clearances of PRV guidance, as well as the clearance between PRV-dome bracket and PRV-dome-support during cold and operating conditions (§3.9.2).	After each refueling outage, prior to start-up, the clearances of the accessible components and their supports are inspected [49]. During the refueling outage of 2012, the clearances of the pressurizer supports and whip restraints were checked as well [53].
Steam generator			
- SG shell and cover - Primary inlet and outlet nozzle - Feed water and main steam nozzle - Tubesheet	(Corrosion) fatigue	Revalidate the existing fatigue analysis, validate assumptions using FAMOS (§3.5) and consider environmental effects (§3.7.8).	These SCs are considered in the TLAA fatigue project [52].

SCs	Relevant ageing mechanism	Recommendation	Implementation
Feed water and main steam nozzles	FAC	Evaluate the feed water and main steam nozzle within the FAC screening analysis (§3.6.9).	Based on the environmental conditions and the applied materials KCB does not expect any significant problems with FAC for the feed water and main steam nozzles. In addition former inspections did not show any FAC degradation. One of the feed water nozzles was inspected on the inside in 1981, when the connected bend was replaced. Both feed water and main steam nozzles were inspected in 1997, when they were accessible due to the replacement of the feed water and main steam lines for leak before break piping [56]. However, to assure that FAC is not a problem during LTO, KCB will evaluate whether FAC can result in significant wall thinning of the feed water and main steam nozzles. A note in this respective is that KCB plans to implement the Condition Oriented Ageing and Plant Life Monitoring System (Comsy), a software tool for the assessment and documentation of NPPs mechanical components in respect to several degradation mechanisms including FAC.
U-tubes (external)	Wastage Pitting corrosion IGC Crevice corrosion Denting ODSCC	Ensure that the presence and height of deposits that have accumulated on top of the tubesheet are measured and determine if sludge removal is needed (§3.6.2, 3.6.4, 3.6.5, 3.6.6, 3.6.10, 3.7.5).	KCBs current inspection methods, including the 3-yearly Eddy current testing of 100% of all, non plugged U-tubes, and 100% of the length of the U-tubes, could be used to determine the level of deposit accumulation on top of the tubesheet. KCBs steam generator workgroup, which contains different disciplines, e.g., chemists, material and inspection specialists, will evaluate whether deposit accumulation should be determined.

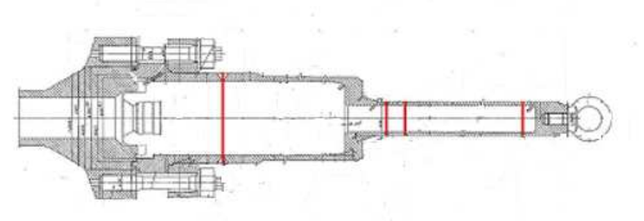
SCs	Relevant ageing mechanism	Recommendation	Implementation
U-tubes (external)	Frictional impact	Perform a one-time inspection at the top of the tubesheet (VT-1) and evaluate is foreign objects will remain in place throughout LTO (§3.3).	As stated in the AMR report, a study showed that foreign objects present in the SGs will remain in place till 2013, not causing significant fretting damage. As KCB examines 100% of all, non plugged U-tubes, and 100% of the length of the U-tubes every 3 years using (multi frequency mix) Eddy current testing, fretting damage on all tubes can already detected nowadays. In addition KCB could perform a visual inspection of the external side of the U-tubes, when the secondary side of the SGs is open. However, as only a small part of the U-tubes is accessible from the secondary side, this inspection can only be performed on a limited number of U-tubes.
Tubesheet	PWSCC	Perform regularly a VT-1 inspection (§3.7.2).	PWSCC has been identified as a relevant ageing mechanism for the welds and cladding of the SG tubesheet in the AMR. KCB will ensure the area of the tube sheets is visually inspected, using SUSI or an equivalent inspection method.
Tubesheet	Pitting corrosion Crevice corrosion	Ensure that the presence and height of deposits that have accumulated on top of the tubesheet are measured and determine if sludge removal is needed (§3.6.4, 3.6.6).	KCBs current inspection methods, including the 3-yearly Eddy current testing of 100% of all, non plugged U-tubes, and 100% of the length of the U-tubes, could be used to determine the level of deposit accumulation on top of the tubesheet. KCBs steam generator workgroup, which contains different disciplines, e.g., chemists, material and inspection specialists, will evaluate whether deposit accumulation should be determined.

SCs	Relevant ageing mechanism	Recommendation	Implementation
Tube plugs (rolled)	Frictional impact	Add a visual inspection of the rolled tube plugs to the ISI plan (§3.3).	Loose parts which could accumulate in low flow areas within the primary and secondary side of the SGs can cause damage to the rolled tube plugs. In response to external operating experience KCB performed a one-time visual inspection of 20 rolled tube plugs in SG1 and 26 rolled tube plugs in SG2 during the refueling outage of 2010. During the 3 yearly Eddy current testing of the U-tubes, KCB visually checks the plugged positions to verify the actual location of the probe. A visual inspection of the rolled tube plugs is however, not explicitly mentioned in any documentation. KCB will incorporate the visual inspection of the tube plugs in the examination procedure and/or ISI plan and will document the findings of this visual inspection.
Tube plugs (welded)	PWSCC	Perform VT-1 inspection when the primary side is open for inspection (§3.7.2).	KCB opens the primary side of the steam generator every 3-years to determine the material condition of the U-tubes using Eddy current testing. During this investigation all accessible components, including the welded tube plugs, on the primary side of the steam generator are visually inspected. The visual inspections and their results (as no indications have been found) are at the moment not explicitly mentioned in any documentation. KCB will document the findings of these visual inspections for future inspections.

SCs	Relevant ageing mechanism	Recommendation	Implementation
Tube supports (incl. comb structure)	Pitting corrosion	Ensure that the presence and height of deposits that have accumulated on top of the tubesheet are measured and determine if sludge removal is needed (§3.6.4).	KCBs current inspection methods, including the 3-yearly Eddy current testing of 100% of all, non plugged U-tubes, and 100% of the length of the U-tubes, could be used to determine the level of deposit accumulation on top of the tubesheet. KCBs steam generator workgroup, which contains different disciplines, e.g., chemists, material and inspection specialists, will evaluate whether deposit accumulation should be determined.
Underclad cracks in the channel head ring and tubesheet of both SGs	-	Evaluate if the underclad cracking evaluations performed for the RPV can be used to exclude that the SG underclad cracks will undergo negligible growth during LTO (§3.9.3).	KCB will evaluate if the SG underclad cracks undergo negligible crack growth during LTO. As the loading conditions and crack/component configuration of the SG underclad cracks are probably bounded by those of the RPV, the RPV underclad crack evaluation can probably be used in this evaluation.
SG support guides and whip restraints	-	Check the lower support clearance between console construction and SG brackets, as well as the clearance of horizontal guidance bar of the lower SG support during cold and operating conditions (§3.9.2).	After each refueling outage, prior to start-up, the clearances of the accessible components and their supports are inspected [49]. During the refueling outage of 2012, the vertical clearances of the SG supports and whip restraints were checked as well [53].
SG supports and welds that attach the support to the SG shell	-	Perform a VT-3 of supports during each ISI inspection interval (§3.9.1).	After each refueling outage, prior to start-up, the accessible components including the SG supports are visually inspected (VT-3) [49].

SCs	Relevant ageing mechanism	Recommendation	Implementation
Main coolant pumps			
All relevant MCP subcomponents	(Corrosion) fatigue	Revalidate the existing fatigue analysis, validate assumptions using FAMOS (§3.5) and consider environmental effects (§3.7.8).	These subcomponents are considered in the TLAA fatigue project, including validation of assumptions of environmental effects [52].
Emergency seal water supply lines	Fatigue	Evaluate fatigue for the emergency seal water supply lines (§3.5).	Based on the loading conditions, as well as the dimensioning of the emergency seal water supply lines KCB does not expect fatigue to be a safety issue. To exclude fatigue of these auxiliary lines KCB will perform a fatigue assessment.
HP cooling circuit	-	Review the current inspection plan for the weld connections in the MCP HP cooling circuit (§3.9.5).	ASME recommends a surface examination of all terminal ends and joints where each pipe or branch run is connected to other components where the stress levels exceeds either a CUF of 0,4 or a certain stress intensity range. As fatigue has been identified as a relevant ageing mechanism for the emergency seal water supply lines, KCB will perform a fatigue assessment for these lines (see above). However KCB does not expect fatigue of these lines will be a safety issue and requires a revision of the ISI scope. In addition ASME requires that 25% of the welds in the HP cooling circuit is examination using surface inspections. KCB follows this AMSE requirement.
MCP supports and welds that attach the support to the MCP casing	-	Perform a surface/volumetric examination of the MCP support welds and a VT-3 inspection of supports during each ISI	As stated in the AMR, KCB last inspected the MCP casing support welds in 1995 (MCP1) and 1998 (MCP2). No further inspections of these welds are scheduled in the ISI plan. In order

SCs	Relevant ageing mechanism	Recommendation	Implementation
		inspection interval (§3.9.1). Check that the MCP support assembly is in accordance with the assembly drawings (§3.9.1).	to ensure that no unexpected ageing mechanisms go unnoticed, KCB will add the UT-inspection of these welds to the ISI plan. After each refueling outage, prior to start-up, the accessible components including the MCP supports are visually inspected (VT-3) [49]. In addition a visual check of the condition of the MCP supports was scheduled in the refueling outage of 2012. No indications were found. [53].
Tension studs of the lower support construction	Concrete shrinkage	Check the pretension of tension studs every 8 years (§3.8.5).	During the refueling outage of 2012 KCB checked the pretension of these tension studs. No indications of loss of pretentions were found [53].
Primary support guides and whip restraints	-	Check the clearances of horizontal whip restraints and the clearance between hold-down device and MCP bracket during cold and operating conditions (§3.9.2).	After each refueling outage, prior to start-up, the clearances of the accessible components and their supports are checked [49][44]. During the refueling outage of 2012, the vertical clearances of the MCP supports and whip restraints were checked as well [53].
Control rod drive mechanism housings			
Potentially all CRDM pressure housings parts	(Corrosion) fatigue	Evaluate if a fatigue analysis of the CRDM pressure housing is necessary for LTO. (§3.5) (§3.7.8).	As stated in the AMR report it is not expected that fatigue is a major issue for the CRDM pressure housing. To assure that fatigue is not an issue KCB will determine if a fatigue analysis should be performed.
CRDM pressure housings	-	Inspection of CRDM pressure housings internal surfaces, as well as its welds. Conform ASME XI or KTA (§3.9.4).	In order to determine the physical status of the CRDM pressure housings a volumetric examination according to ASME XI (using UT) of the CRDM pressure housing welds is prepared.

SCs	Relevant ageing mechanism	Recommendation	Implementation
			Each CRDM pressure housing contains 4 applicable welds (see figure below). In addition a wall thickness measurement at the upper part of the pressure housing (where the boundary is of water an trapped air during operation) is prepared [54].  These inspections of CRDM pressure housings are planned for the outage of 2013.
Main coolant lines and pressurizer surge line			
All MCL and pressurizer surge lines components and subcomponents	(Corrosion) fatigue	Revalidate the existing fatigue analysis, validate assumptions using FAMOS (§3.5) and consider environmental effects (§3.7.8).	These subcomponents are considered in the TLAA fatigue project, including validation of assumptions of environmental effects [52].
Surge line and the surge line nozzle at the MCL	Thermal fatigue	Evaluate the current loading configurations during normal operation (§3.5).	In 2010 fatigue monitoring system (FAMOS) instrumentation was installed and implemented at KCB. The FAMOS system monitors thermal loading, in several fatigue relevant locations on the primary circuit and several points on the main feed water system piping. It is expected that KCB has gathered sufficient measurement information by the end of 2015 to calculate representative thermal loads on the surge line. In this process KCB will assess if there is any need to revise the existing

SCs	Relevant ageing mechanism	Recommendation	Implementation
			operating procedures.
Steel containment			
Carbon steel or low alloy steel high tensile bolting	HISCC	Perform a visual inspection and check of the strength of high tensile bolt connections (§3.7.6).	Based on the controlled environment inside the steel containment KCB does not expect any problems due to HISCC of the steel containment high tensile bolt connections. Nevertheless KCB will perform a one-time visual inspection of the accessible bolt connections.
Reactor pressure vessel internals			
Baffle-former bolts	IASCC	Verify the expected fluence over the cycles 15-60 to determine if and when IASCC could become a possible ageing mechanism for the baffle-former bolts (§3.7.4).	To determine if and when IASCC can become a possible relevant ageing mechanism, KCB will evaluate the expected fluence for the baffle-former bolts over cycles 15 – 60. The existing axial fluence calculations, which calculated the expected fluence from cycle 1 till cycle 40 and 60, will be used as the basis for this evaluation.
Secondary systems			
RL040/050Z101 RL040/050Z003 RS011/021Z003 RA001/002Z102 RA001/002Z002	(Corrosion) fatigue	Revalidate the existing fatigue analysis, validate assumptions using FAMOS (§3.5) and consider environmental effects (§3.7.8).	These forgings and nozzles are considered in the TLAA fatigue project, including validation of assumptions of environmental effects [52].

SCs	Relevant ageing mechanism	Recommendation	Implementation
Safety related auxiliary systems			
Tubesheets of TA000B001/002	(Corrosion) fatigue	Revalidate the existing fatigue analysis, validate assumptions using FAMOS (§3.5) and consider environmental effects (§3.7.8).	TA000B001/002 s are considered in the TLAA fatigue project, including validation of assumptions of environmental effects [52].
Polypropylene piping TR010Z004/005/006/007	Ageing of thermoplastic materials	Evaluate the degree of material degradation and determine if replacement is necessary (§3.8.4).	KCB will either evaluate the degree of material deterioration and based on the results determine whether these piping sections need to be replaced, or replace these piping sections on beforehand.
Mechanical fasteners			
Carbon steel, low alloy or martensitic stainless[7] steel high strength fasteners in an aqueous environment	HISCC	Verify that leakages of all in-scope fasteners will be noticed using the leakage monitoring system and/or plant walk downs (§3.7.6). Replace austenitic stainless steel fasteners of steel quality A2 and A3 by fasteners made of A4 or A5 in case the presence of chlorides cannot be excluded (§3.7.6).	KCB performs evaluations of existing activities (e.g., plant walk downs) and corresponding procedures. The purpose of this process is to evaluate the existing activities and procedures with regard to any improvement thereof. The use of austenitic stainless steel fasteners of steel quality A2 and A3 in an environment where the presence of chlorides cannot be excluded is not conform KCBs engineering standard. Therefore, KCB does not expect that these fasteners are installed in such environment. In the case some fasteners would be installed in such environment, HISCC could result in leakages,

[7] In the original design martensitic steel fasteners were not used for the nuclear components. During project "Modificaties" in 1997, martensitic steel fasteners were introduced from the French specification for some Pressurizer parts.

SCs	Relevant ageing mechanism	Recommendation	Implementation
			which can be identified using KCBs leakage monitoring systems and during the regular plant walkdowns. In order to assure that austenitic stainless steel fasteners of steel quality A2 and A3 will not be installed in the future, KCB will check its stock inventory and their order specifications.

Table A-2 Electrical and I&C AMR

Recommendation	Implementation
General	
Incorporate visual inspections in all relevant inspection documentation.	KCB will incorporate the performance of visual inspections in all relevant maintenance instructions ("werkomschrijvingen").
Cables	
Extent the visual inspections, executed when ether maintenance on an active component is performed, to cables in the corresponding function chain exposed to high temperatures and/or dose rates.	KCB will evaluate whether the extent of visual inspections performed during maintenance activities should be broadened.
Implement an ageing management program for the cables identified as susceptible to ageing.	Cables exposed to high level temperatures or dose rates (hot spots) are included in a replacement program. Within this program the ageing condition of the removed cables is examined [50]. In addition KCB will evaluate the environmental conditions of the cables identified in the AMR as susceptible to ageing (several PVC/PVC, PE/PVC and SIR/SIR cables). Based on this evaluation KCB will decide whether additional measures are necessary.
Implement an ageing management program for medium voltage PVC cables	Operating experience has shown an enhanced failure rate of PVC medium voltage cables (breakdown of the cables and decreased insulation resistance). KCB will evaluate whether additional measures for these cables are necessary.
Wires	
Implement an ageing management program for wires in the I&C cabinets.	KCB will evaluate whether a separate ageing management program for wires in the I&C cabinets is necessary.
Electrical connections	
Determine all electrical connections inside the containment and investigate	

Recommendation	Implementation
whether the applied materials are exposed to large amounts of radiation.	
Extent the visual inspections, executed when ether maintenance on an active component is performed, to connectors and in particular terminals of the corresponding function chain.	KCB will evaluate whether the extent of visual inspections performed during maintenance activities should be broadened.
Investigate whether connector and terminal materials EP, FKM, PA, SIR or SEBS are installed inside the containment or in other high level radiation areas.	KCB will investigate during the next refueling outage whether these materials are applied for electrical connections in junction boxes. If more insight in the radiation level in a specific area is needed to determine whether an electrical connection is susceptible to ageing, radiation monitors will be installed. These monitors will record the radiation levels during one cycle. Based on these measurements KCB will evaluate if replacements are necessary[55].
Implement an ageing management program for electrical connections; including cable junction sleeves, connection terminals, swivels and soldered connections.	KCB will investigate whether additional ageing management activities for these components are necessary.
Electrical containment penetrations	
Investigate whether the materials FKM and SIR are installed on the containment side of the electrical penetrations.	KCB will investigate whether these materials are installed on the containment side of the electrical penetrations.
Implement an ageing management program for the conductor materials Cu, Cu-alloy and Ni-Fe-alloy installed in electrical penetrations in a moist environment.	KCB will evaluate whether a separate ageing management program for these conductor materials installed in electrical penetrations in a moist environment is necessary.
Perform a visual inspection of the electrical insulation when ether a He-leakages tests is performed.	KCB will evaluate whether visual inspections of the insulation of electrical containment penetrations is necessary and possible.
Cable penetrations (floors, walls)	
Perform a visual inspection of the cable penetrations every five years.	KCB will evaluate whether additional visual inspections of the cable penetrations is necessary.

Recommendation	Implementation
Cabinets and racks	
Perform a visual inspection of the cabinets and racks every five years.	KCB will evaluate whether additional visual inspections of the cabinets and racks is necessary.
Cable trays	
Perform a visual inspection of the cable trays every five years.	KCB will evaluate whether additional visual inspections of the cable trays is necessary.
Check the mechanical loads of the cable trays.	KCB will evaluate whether a check of the loads on the cable trays is necessary.