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1.0 Executive Summary

Koeberg Nuclear Power Station (Koeberg) has produced clean, safe, reliable electrical power for more than 39 years. Its operating licence is valid for 40 years, and approval by the National Nuclear Regulator (NNR) is required for it to continue to operate beyond the 40-year licensing period. This safety case has been produced in support of the application for long-term operation (LTO) and demonstrates that the regulatory requirements for LTO are met and that it is safe to continue operating for an additional 20 years, from 2024 to 2044 (Unit 1) and 2045 (Unit 2).

The national nuclear regulatory framework for LTO is provided by the NNR Act and the LTO regulations, with additional guidance given in the regulatory guide. The requirements to be demonstrated by the safety case are clearly defined in the LTO regulations, among others:

- demonstrate compliance with relevant regulatory safety criteria and requirements;
- base the application on the results of safety analysis, with due consideration of the ageing of structures, systems, and components (SSCs);
- provide an overall assessment of the safety of the nuclear installation and justification for continued safe operation;
- demonstrate the availability of financial and human resources as well as knowledge management; and
- include the necessary safety improvements in the application, including refurbishment, provision of additional SSCs, safety analyses, and engineering justifications, to ensure that the licensing basis remains valid during the LTO period.

Several assessments have been conducted to support LTO. A feasibility study was undertaken to determine the plant modifications necessary for plant life extension. A periodic safety review (PSR) was conducted to obtain an overall view of plant safety and determine safety improvements needed for continued safe operation. An assessment of safety aspects of long-term operation (SALTO) was conducted to review the effectiveness and completeness of Koeberg's ageing management programmes and implement improvements to ensure that equipment ageing was adequately managed. The specific site characteristics were reviewed to confirm the suitability for nuclear siting, considering the latest methodologies and operating experience. These assessments, among others, are drawn on throughout the safety case to support the arguments and provide evidence.

The regulations on safety standards and regulatory practices specify the regulatory requirements, including the principal safety criteria, applicable to holders of nuclear authorisations. The PSR review of the probabilistic safety assessment (PSA) confirmed that the principal safety criteria are respected and that a framework exists in the form of processes and procedures to ensure that the criteria would be respected during the LTO period. Although the risk profile had shown an increase due to recent cask operations, the PSA results demonstrated that the risk associated with operations of the plant is well within the principal safety criteria (that is, risk limits) specified in RD-0024 and, therefore, supports continued safe operation. Koeberg also complies with the dose limits and dose constraints specified in RD-0024 and RD-0022.

The PSR review determined that the current licensing basis remains valid; however, the Koeberg Site Safety Report (KSSR) was deemed to be outdated, and gaps with regard to NNR requirements were identified. Safety improvements (site characterisation assessments) to address these deviations are included in the PSR integrated implementation plan (IIP) and prioritised to be resolved prior to entry into LTO. The site characterisation assessments identified changes in the hazards relating to seismic activity, flooding with tsunami, coastline erosion, and wind speeds. Coastline erosion and wind speeds were found to be low-risk hazards to the Koeberg site. The tsunami assessment revealed that the probable maximum tsunami (PMT) run-up and inundation were governed by volcanic flank collapse tsunamis. Further assessment to gain an understanding of the potential impact on the plant was conducted and concluded that the probability of such a tsunami is low and mitigations exist to ensure the safety of the plant. A more detailed analysis to confirm the preliminary findings is included in the LTO integrated preparation plan (IPP), pre-LTO activities, and scheduled for completion prior to LTO.

Relating to the seismic hazard, Senior Seismic Hazard Assessment Committee (SSHAC) studies are ongoing and will be completed prior to LTO. An interim seismic evaluation based on the Electric Power Research Institute (EPRI) procedure, referred to as the expedited seismic evaluation process (ESEP), was conducted. This evaluation provided reasonable assurance that the Koeberg units are sufficiently robust to shut down safely and cope with a significant seismic event and loss of alternating current (AC) power with few enhancements to be implemented as recommended by the evaluation. These activities are included in the LTO IPP.

The current plant design was assessed during the PSR plant design review to determine the adequacy of Koeberg's design (including design documentation) against the current licensing basis and national and international standards. It was found that the plant design largely complies with regulatory requirements and that the design basis conforms to modern national and international safety standards, codes, and practices. Although a deviation associated with control room habitability not meeting the dose criterion for accidents was graded as high during the PSR, mitigations have since been implemented to ensure that the dose criterion is met.

Regarding the design management system, processes and procedures are in place to fulfil the requirements of a design management system (including a configuration management system) and ensure that sufficient, accurate information, consistent with the physical plant and operational characteristics, is available in a timely manner. The processes and procedures are broadly aligned with international good practices and ensure that plant design changes are effectively controlled and are fit for purpose.

PSR confirmed that the application of defence in depth (DiD) is embedded in plant design, operations, and management systems. It had been demonstrated that the deviations identified have no more than minimal impact on DiD, as adequate compensatory measures exist, and there is no significant cumulative effect. There is adequate independence between the individual levels of defence. The fundamental safety functions (FSFs) are ensured despite the impact of the deviations, as they do not have a cumulative effect on the FSFs and are mainly of a low safety significance. Protection of the physical barriers, namely, fuel cladding, the reactor coolant system, and the containment, which is integral to an effective DiD, is extensively considered in the Safety Analysis Report (SAR), with available design provisions to ensure the protection of the barriers. Safety improvements, particularly related to design extension conditions (DECs – Level 4) such as hardened water supply and water connections, were recognised, will further enhance Koeberg's DiD, and are included in the PSR and LTO IIPs.

The International Atomic Energy Agency (IAEA) safety standards and guidelines identify events that are worthy of practical elimination, due to their significance. Koeberg has provisions in place for the practical elimination of these events. While practical elimination is not achieved in all cases, the levels of DiD and provisions available are commendable and will be further enhanced with the implementation of safety improvements in the LTO IIP.

The plant life extension feasibility study identified major components such as steam generators, reactor vessel heads, and refuelling water storage tanks for replacement to support LTO. The refuelling water tanks and the reactor vessel heads have since been replaced, and the steam generator replacements are scheduled for completion prior to LTO.

A comprehensive ageing management evaluation was performed to determine whether the ageing mechanisms of SSCs important to safety were managed in a manner that would ensure that they continued to fulfil their intended design function with an adequate safety margin for the entire period of LTO. Ageing management programmes (AMP) for all the in-scope SSCs were evaluated against the nine attributes of an effective AMP. The evaluation identified the corrective actions and safety improvements required to be implemented for safe LTO. These have been prioritised and scheduled in accordance with the LTO IIP.

Ageing management is effectively integrated into Koeberg's design change processes to ensure that ageing mechanisms are adequately considered in the design stage and all subsequent stages of the plant life cycle.

The LTO assessment identified some SSCs important to safety with ageing mechanisms that posed a risk if not treated in a timely manner, namely, containment buildings, aseismic bearings, cables, and switchboards. The containment buildings are subject to chloride-induced reinforcement corrosion. The proposed solution is to implement an impressed current cathodic protection (ICCP) system into the concrete of the containment buildings to neutralise the corrosion effects of chlorides. The containment buildings are acceptable for operation at present based on current surveillance monitoring results. An integrated leak rate test (ILRT) was completed in 2015 (on both units), and the containment buildings' safety analysis (time-limited ageing analysis) determined that the structural integrity of the containment buildings was ensured for the planned LTO period. The ILRT will continue to be conducted in line with the requirements of the ageing management programme for the buildings. The ICCP modification and the next scheduled ILRT are included in the LTO Implementation Plan (IP).

Koeberg has monitored the seismic bearings according to the monitoring programme in terms of regulatory requirements, and all results and findings have been submitted to the NNR. Only limited maintenance (plate interface seal replacement) had been required. The visual inspection and test results of the in-situ and sample bearings have shown no significant change in material properties. An evaluation to better characterise the bearings has been completed which confirmed that the ageing mechanisms and effects are understood and are managed effectively. Based on these facts, the bearings can perform their design function and remain fit for purpose and suitable for LTO.

Large-scale switchboard and cable replacements are not anticipated based on industry practice, and the failure rate is low. Availability of major spares such as circuit breaker and contactor modules for both the 6,6 kV and 380 V switchboards has been confirmed. All in-scope plant electrical switchboards, switchboard components, and plant cabling are comprehensively addressed and aligned with the IAEA International Generic Ageing Lessons Learned (IGALL) requirements. The service life of equipment qualification (EQ) cables was confirmed for the full period of LTO based on the outcomes of the time-limited ageing re-analysis).

An ageing management assessment was also conducted on the SSCs used to support other licence-binding programmes. These licence-binding programmes are radiation protection, emergency planning, environmental monitoring, chemistry, licenced operator training, and nuclear security (The nuclear security report is a separate and confidential submission). It was found that the ageing management of these SSCs is governed by procedures that ensure the long-term reliability of the equipment. Continued reliability will be ensured during LTO through maintenance, inspection, and testing activities.

A suite of ageing management procedures for AMP reviews, scope setting, and time-limited ageing analyses (TLAAs) has been developed and is embedded within the Koeberg integrated management system, including the Koeberg licencing basis manual (KLBM). This ensures that ageing management will continue to be systematically implemented throughout the period of LTO.

One of the major aspects of a nuclear power operation is the management of radiation exposure with the goal of minimizing the harmful effects of ionizing radiation. Therefore, the scope and effectiveness of the radiation protection programme was assessed during the PSR safety performance review. The review concluded that there are adequate processes, procedures, guides, and work instructions and that these are adequate and effective to support LTO.

Koeberg assessed the prospective dose to the public and the environment using a conservative source term. The results obtained considering environmental build-up historically and for the next 20 years showed that the dose to a member of the critical group/representative person is estimated to be around 94 µSv/a. This assessment of the prospective cumulative dose, considering planned activities at the Duynefontyn site, is below the dose constraint of 250 µSv/a as stipulated in the regulations on safety standards and regulatory practices, R.388, and the dose limit for Koeberg as stipulated in RD-0022.

The results of the ecological risk assessment for marine and terrestrial reference organisms in the Duynefontyn site safety report (DSSR) concluded that the liquid and gaseous discharges from the facility were unlikely to pose a significant risk to the environment. Although the NNR has not issued a regulatory limit on non-human biota, the results showed that the dose rate for both terrestrial and marine environments was well below the dose rate guideline values as applied internationally (that is, 10 µGy/h for marine organisms and 40 µGy/h for terrestrial organisms).

Regarding the adequacy of the emergency preparedness and response, the PSR on emergency planning concluded that the current emergency preparedness and response plan is adequate for LTO. An integrated Koeberg nuclear emergency preparedness and response plan (IKNEP) had been documented, and exercises with interfacing organisations (NNR, local and provincial government, and international partners) were held regularly. The IKNEP was informed by a comprehensive safety analysis undertaken to ascertain all exposure sources and evaluate radiation doses associated with the facility as contained in the emergency planning technical basis (EPTB). The DSSR did not find any factors around the site that could impede the implementation of the emergency plan. The EPTB was reassessed based on the requirements in RD-0014, and it was concluded that the current emergency zones remain valid. During the LTO period, Koeberg will continue to ensure that proposed developments around the site do not impede the emergency preparedness and response plan.

The feasibility of LTO includes the consideration of the adequacy of waste management strategies. The effectiveness of the radioactive waste management programmes was reviewed during the PSR safety performance review, and compliance with regulatory requirements was confirmed. This review assessed whether Koeberg had implemented programmes for the minimisation and safe management of radioactive waste at the site. Although additional radioactive waste will be generated as part of normal operations during the LTO period, the generated volume of waste will remain largely unchanged; thus, the current radioactive waste management regime and strategy remain effective and adequate for safe LTO.

Vaalputs will remain the disposal facility for low- and intermediate-level waste – short-lived and have sufficient capacity to accommodate waste generated during the LTO period. This was confirmed with the National Radioactive Waste Disposal Institute (NRWDI).

Spent fuel is currently stored in two spent fuel pools and in dry storage casks on site. The Koeberg spent fuel strategy caters to the transfer of spent nuclear fuel from the spent fuel pools into the dry storage casks in the cask storage building (CSB). In line with this strategy, the transient interim storage facility (TISF) is being constructed to store the dry casks, while the government establishes the centralised interim storage facility (CISF). Eskom will ensure that spent nuclear fuel is safely stored in the TISF while the CISF is being constructed.

The potential impact of LTO on the environment was assessed in the PSR and confirmed to be insignificant. The results of the assessment showed that the effluent and environmental monitoring programmes ensured that emissions and discharges were properly controlled and were as low as reasonably achievable and that no changes to these programmes were required for LTO. It was demonstrated that operational practices to ensure environmental protection met NNR requirements, conformed to relevant international guidelines, and ensured safe operation.

Organisational provisions associated with human resources, financial provisions, and knowledge management were assessed against national and international requirements to confirm their adequacy for the safe operation of the plant for LTO. The review concluded that adequate organisational provisions are available to support LTO.

The human resource requirements are reviewed annually using a workforce planning process. The process utilises a 10-year planning window, thus allowing long-term resource needs to be identified. The human resources strategy utilises a combination of permanent staff, fixed-term contracts, and outsourcing of services to ensure that a sufficiently skilled workforce is available for safe operation. Eskom has embarked on a recruitment campaign to ensure adequate staff for LTO.

A knowledge management process has been established to ensure knowledge capturing and transfer throughout the period of operation. The knowledge management process uses an integrated approach to identifying, capturing, evaluating, retrieving, and sharing relevant information assets (such as databases, documents, policies, procedures, and previously uncaptured expertise and experiences from individual workers). The process will be implemented in all departments within the nuclear operating unit (NOU) before LTO and is included in the LTO IIP.

Major expenditure for Koeberg is associated with salaries (operational cost) and safety improvements in the nuclear technical plan (capital costs). The LTO IPP has been sufficiently resourced and funded. A skills, time, and cost analysis were conducted for the PSR IIP and concluded that the cost requirements for the PSR IIP were in line with past approved expenditures for a similar scope of activities. Eskom has adequately funded the Koeberg operational costs over the past 39 years. Therefore, financial resources for LTO are available for operational costs and the LTO scope of activities.

The safety improvements resulting from the LTO assessments (including the safety aspects of long-term operation (SALTO), the PSR, and the DSSR) have been identified and scheduled in the LTO IIP. The safety improvements are categorised into two groups, namely, the LTO integrated preparation plan (which are safety improvements required prior to entry into LTO) and the LTO implementation plan (which are safety improvements that will be implemented during the LTO period).

An overall assessment of the safety of Koeberg for LTO has been provided by the safety case, which draws on the outcomes of the LTO assessments. The requirements for safe LTO as documented in the LTO regulations R.266, RG-0027, and NIL-01 Variation 19 have been met. the safety improvements associated with ensuring the safe operation of the plant have been provided in line with the requirements of the regulations. No safety concerns were identified during the LTO assessments that would preclude the plant from entering LTO, and as confirmed by the PSR, Koeberg is safe to continue operations into LTO. It has been demonstrated that nuclear safety will be maintained in accordance with the licensing basis and international good practices for the intended period of LTO, with the timely implementation of safety improvements contained in the LTO IIP.

2.0 Introduction

Globally, Long Term Operation (LTO) has become a common practice within the nuclear industry. More than 130 nuclear reactors worldwide (representing more than 30% of the global nuclear fleet) are operating beyond their initial 40-year licensed periods. As the average age of the world's nuclear fleet is greater than 30 years, it is expected that the role of LTO will continue to expand in the next decade. In many countries, LTO presents the least-cost option for delivering safe and reliable baseload power generation capacity as nations seek to reduce their carbon emissions in accordance with their nationally determined carbon emission targets.

The Koeberg Nuclear Power Station (Koeberg) is the only nuclear power plant on the African continent and is wholly owned and operated by Eskom. It is located 30 km from Cape Town near Melkbosstrand on the West Coast of South Africa. The plant was originally licenced to operate for 40 years. Construction of the plant commenced in 1976, and the plant started commercial operation with Unit 1 in July 1984, followed by Unit 2 in November 1985. The nuclear installation licence (NIL-01 Variation 19), amended in 2019, stipulates the conditions of operation for the facility. It is valid until 21 July 2024, unless amended for subsequent licensing stages, including LTO, or varied, suspended, or revoked. Therefore, with the LTO regulations, a licence approval for the additional years of operation is required from the NNR to continue to operate. Eskom has decided to apply for a further 20 years of operation.

In addition to Variation 19 of the licence in 2019, the NNR also issued two regulatory guides stipulating requirements for LTO. RG-0027 (*Ageing Management and Long-Term Operation of Nuclear Power Plants*) [294] prescribes the requirements for extended commercial power operations, including requirements for ageing management and the demonstration of achievement of regulatory requirements during the LTO period. RG-0028 (*Periodic Safety Review for Nuclear Power Plants*) [295] prescribes the requirements for performing a periodic safety review (PSR) in support of LTO.

Therefore, the safety case aims to demonstrate compliance with the LTO regulations. It provides an overall conclusion on the safety of Koeberg for the intended 20-year period of LTO. It discusses and evaluates the adequacy of the plant design, the adequacy and effectiveness of safety-related programmes, and organisational provisions, especially ageing management programmes. The main inputs to the safety case are the PSR and the Safety Aspects of Long-Term Operation (SALTO) ageing management assessments; and where applicable, other relevant safety-related assessments were performed to complement the case for continued safe operation.

3.0 Applicability

This document applies to the Nuclear Operating Unit (NOU), including Units 1, 2 and 9 of Koeberg, and other functional areas that are covered by the LTO assessment programme, such as nuclear security, emergency planning, radiological impact on the environment and human factors.

4.0 Supporting Clauses

4.1 Definitions

Term	Definition
1E	Safety-related electrical equipment qualified according to IEEE standards.
Affaire Parc	A generic issue affecting the EDF PWRs and analysed by multi-disciplinary teams of experts.
Ageing management	The process whereby analyses, tests, and assessments are performed to determine whether the environment and service conditions can cause or have caused the ageing of SSCs.
Ageing management programme (AMP)	A set of policies, processes, procedures, arrangements and activities for managing the ageing of SSCs for a nuclear power plant.
Cable ageing	A process induced by adverse localised environments and service conditions, resulting in ageing effects.
Current licensing basis	The safety case applicable at any time during the operation of the plant, comprising applicable regulations and Regulator guidelines and all licence-binding documents, including project management documents, safety analysis report, operational limits and conditions, and other safety-related programmes applicable during a licensing stage (including modifications), which shall be retained as records.
Casking Operations	Cask operations involve loading or unloading spent fuel assemblies underwater (or in a hot cell) to move from one place to another for storage. The loaded cask or canister is sealed and dried, as required for containment of the spent fuel, followed by various subsidiary operations by means of ancillary systems.
Decommissioning	The process by which nuclear power stations or installations are retired from service. This process occurs when nuclear facilities have reached the end of their useful life, are no longer economically viable to operate, or when the operating licence has been irrevocably withdrawn.
Design Extension Conditions (DEC)	Postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process for the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits.
DEC-A	Design extension conditions in events without significant fuel degradation.
DEC-B	Design extension conditions in events with core melting.

Term	Definition
Development	Any construction, utilisation of land, zoning or rezoning of land or the subdivision thereof, other than the predominant or permitted use in terms of existing applicable zoning schemes and includes any new enhancement of a right of use by way of a departure, an application for a guest accommodation or second dwelling unit, conditional or consent use, rezoning of land, the subdivision thereof, additional rights of use or changes to the relevant restrictions imposed by any zoning scheme.
Deviation	A negative finding to the national regulations, the Koeberg licence, international requirements, or those codes, standards, processes, or practices adopted to meet those requirements.
Dose	The amount of radiation received, where a more specific term such as 'effective dose' or 'equivalent dose' is not necessary for defining the quantity of interest.
Dose constraint	A prospective and source-related restriction on the individual dose arising from the predicted operation of the authorised action serves exclusively as a bound on the optimisation of radiation protection and nuclear safety to limit the range of options considered in the optimisation process and to restrict the doses via all exposure pathways to the average member of the critical group to ensure that the sum of the doses received by that individual from all controlled sources remains within the dose limit, and which, if found retrospectively to have been exceeded, should not be regarded as an infringement of regulatory requirements but rather as a call for the reassessment of the optimisation of radiation protection.
Dose limit	The value of effective dose or equivalent dose to individuals from actions authorised by a nuclear installation licence, nuclear vessel licence, or registration certificate must not be exceeded.
Emergency planning	The process of developing and maintaining the capability to take actions to mitigate the impact of an emergency on persons, property, or the environment [148].
Emergency preparedness	The capability to promptly take actions that will effectively mitigate the impact of an emergency on persons, property, or the environment [148].
Emergency response	The performance of actions to mitigate the impact of an emergency on persons, property, or the environment [148].
Environmental conditions	The ambient physical states that surround an SSC (for example, temperature, radiation, humidity).
Evacuation	Urgent removal of a population from an area to avoid the imminent or actual threat of acute radiation exposure, for example, from an airborne radioactive plume.
Evacuation time	In the precautionary action planning zone (PAZ), means four hours from the time that an evacuation order is given and the urgent protective action planning zone (UPZ), means 16 hours from the time that the evacuation order is given.
Exposure	The act or condition of being subject to irradiation.
Finding	Information discovered as the result of the periodic safety review; in the context of the PSR can either indicate a negative finding (gap) with a national or international requirement or a positive finding (strength) where the requirement is exceeded.

Term	Definition
Global issues	A common higher level, cross-functional, underlying issue that results in multiple deviations, potentially in different PSR Safety Factors.
Infrastructure	All amenities, facilities, and services necessary to implement the Koeberg nuclear emergency plan include public communication, protection of the environment and property, transport, personnel, radiation monitoring, decontamination, care of the masses, and medical care.
Intervention level	The level of avertable dose at which a specific protective action or remedial action is taken in an emergency exposure situation or chronic exposure situation.
Long-term operation	Operation of the plant beyond an established time frame set forth by, for example, licence term, design, standards, licence or regulations, which has been justified by safety assessment, with consideration given to life-limiting processes and features of SSCs.
Observation	Observations largely represent isolated incidents of errors or omissions that generally, are of very low or no safety significance; alternatively, they represent potential Areas for Improvement. They do not constitute deviations but represent process, procedural, and organisational Opportunities for Improvement (OFI) or enhancements. Condition Reports (CRs) have been raised to allow the OFIs to be assessed as part of the KNPS Corrective Action Programme, or in the case of improvements to the PSA are tracked as change notices in the PSA database.
Occupational exposure	All exposures to radiation incurred by workers in the course of their work.
Periodic safety review	A systematic reassessment of the safety of an existing facility (or activity) carried out at regular intervals to deal with the cumulative effects of ageing, modifications, operating experience, technical developments, and siting aspects aimed at ensuring a high level of safety throughout the service life of the facility (or activity).
Physical Ageing	Ageing of SSC’s due to physical, chemical and /or processes (ageing mechanisms).
Public exposure	Exposure incurred by members of the public from radiation sources.
Radiation	Ionising radiation.
Radiation protection	The protection of people from the effects of exposure to ionising radiation and the means for achieving this.
Radiation workers	Persons potentially exposed to radiation by virtue of their occupation to more than 1 mSv per annum.
Relocation	The non-urgent removal or extended exclusion of people from a contaminated area to avoid chronic radiation exposure following the passage of a radioactive plume [148].
Risk	Qualitatively expressed, the probability of a specified health effect occurring in a person or group as a result of exposure to radiation or (quantitatively expressed) a multi-attribute quantity expressing hazard, danger, or chance of harmful or injurious consequences associated with actual or potential exposures relating to quantities such as the probability that specific deleterious consequences may arise and the magnitude and character of such consequences.

Term	Definition
Safety aspects of long-term operation	The IAEA peer-reviewed safety assessment focused on the safety aspects of long-term operation for nuclear power plants planning on extending plant life into long-term operation.
Safety factor	A review area is defined in RG-0028, which is assessed in a periodic safety review (PSR).
Sheltering	The protection from exposure to radioactivity may be afforded by buildings of a permanent nature.
Surveillance	Observation or measurement of a condition or functional indicator to verify whether that SSC functions within acceptance criteria.
Technological obsolescence	The lack of spare parts, technical support, suppliers or industrial capabilities.
Thermal ageing	A degradation mechanism that results in embrittlement, loss of structure toughness, hardening, and loss of strength of elastomers.
Traffic evacuation model	An assessment tool approved by the municipal council to assess local traffic patterns and to demonstrate compliance with the Koeberg nuclear emergency plan.
Transient population	Persons who are not permanent residents.
Unit 9	Common plant equipment is available for units 1 and 2.

4.2 Abbreviations

Abbreviation/ Trigram	Definition
AADQ	Annual Authorised Discharge Quantity
AC	Alternating Current
ALARA	As Low as Reasonably Achievable
AM	Ageing Management
AME	Ageing Management Evaluation
AMM	Activity Migration Model
AMP	Ageing Management Programme
AMR	Ageing Management Review
aMSL	Above Mean Sea Level
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
APG	Steam Generator Blowdown System
ARE	Main Feedwater System
ASG	Auxiliary Feedwater System
ASME	American Society of Mechanical Engineers

Abbreviation/ Trigram	Definition
BATNEEC	Best Available Technology not Entailing Excessive Cost
C&I	Control and Instrumentation
CAMP	Cable Ageing Management Programme
CAP	Corrective Action Programme
CC	Concrete Components (ASME classification of subcomponents of the containment structure)
CDF	Core Damage Frequency
CISF	Central Interim Storage Facility
CoCT	City of Cape Town
CR	Condition Report
CRDM	Control Rod Drive Mechanism
CRE	Control Room Envelope
CRF	Circulating Water System
CSB	Cask Storage Building
CUF	Cumulative Usage Factor
DBA	Design Basis Accident
DBE	Design Basis Event
DC	Direct Current
DCF	Dose Conversion Factor
DEC	Design Extension Condition
DiD	Defence in Depth
DMRE	Department of Mineral Resources and Energy
DSA	Deterministic Safety Analysis
DSE	Dossier de Système Élémentaire
DSSR	Duynefontyn Site Safety Report
DVN	Nuclear Auxiliary Building Ventilation System
EAF	Environmentally Assisted Fatigue
EAS	Containment Spray System
ECC	Emergency Control Centre
ECMP	Environmental Condition Monitoring Programme
EDF	Électricité de France
EERI	External Event Response Initiative
EE-SRA	External Events Safety Reassessment
EF	Enhanced Fujita (Scale)

Abbreviation/ Trigram	Definition
EP	Emergency Plan
EPD	Electronic Personal Dosemeter
EPP	Personnel Airlock and Equipment Hatch
EPR	Ethylene Propylene Rubber
EPRI	Electric Power Research Institute
EPSOC	The Emergency Planning, Steering, and Oversight Committee
EPTB	Emergency Planning Technical Basis
EPZ	Emergency Planning Zone
EQ	Equipment Qualification
EQML	Equipment Qualification Master List
EQMM	Equipment Qualification Maintenance Manual
ER	Equipment Reliability
ERA	Execution Release Approval
ERCR	Equipment Reliability Change Request
ESEP	Expedited Seismic Evaluation Process
ETE	Evacuation Time Estimate
FAC	Flow-Accelerated Corrosion
FAD	Airspace Danger Area
FAR	Koeberg Nuclear Power Station's Restricted Airspace
FSF	Fundamental Safety Function
FSP	Fundamental Safety Principle
GCT	Turbine Bypass System
GI	Global Issue
GOR	General Operating Rule
HELB	High-Energy Line Break
HFE	Human Factors Engineering
Hz	Hertz
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
ICCP	Impressed Current Cathodic Protection
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
IER	Industry Event Report

Abbreviation/ Trigram	Definition
IGALL	International Generic Ageing Lessons Learned
IIP	Integrated Implementation Plan
IKNEP	Integrated Koeberg Nuclear Emergency Preparedness and Response Plan
ILRT	Integrated Leak Rate Test
IMS	Integrated Management System
INPO	Institute of Nuclear Power Operations
INSAG	International Nuclear Safety Advisory Group
IPP	Integrated Preparation Plan
ISI	In-Service Inspection
ISIP	In-Service Inspection Programme
ISIPRM	In-Service Inspection Programme Requirements Manual
ISTPRM	In-Service Testing Programme Requirements Manual
JPP	Firefighting Water Production System
JPS	Mobile Fire Protection Equipment
KER	Monitoring and Discharge of Nuclear Island Liquid Radwaste
KIT	Computer Data Processing and Monitoring System
KLBM	Koeberg Licensing Basis Manual
KLLF	Koeberg Licensing and Liaison Forum
KM	Knowledge Management
KNPS	Koeberg Nuclear Power Station
KORC	Koeberg Operations Review Committee
KPI	Key Performance Indicator
KRT	Plant Radiation Monitoring System
KSSR	Koeberg Site Safety Report
kV	Kilovolt
Lai	230 V DC Production and Distribution
LBi	125 V DC Production and Distribution
LCi	48 V DC Production and Distribution
LDA	30 V DC Production and Distribution for Analog Control Supply
LDRB	Low Damping Rubber Bearing
LERF	Large Early Release Frequency
LGi	LGA/D – 6,6 kV Unit Switchboards, LGB/C – 6,6 kV Service Switchboards, LGE/LGF – 6,6 kV Station Switchboards, LGI – 6,6 kV Common Switchboard
LGR	Supply via the Auxiliary Grid Transformer

Abbreviation/ Trigram	Definition
LHi	LHA/B/C – 6,6 kV Safeguard Diesel-Backed Switchboards
LKi	LKA/B/C/D/E/F/G/H/J/L/S and LKM/I/K/N/P/X/Z – Non-Safeguard 380 V AC Switchboards
LILW-SL	Low- and Intermediate-Level Waste – Short-Lived
LLi	LLA/B/C/D/E/G/H/I/J/N/O – 380 V Safeguard Diesel-Backed Switchboards
LLWB	Low-Level Waste Building
LMA	Short-Break 220 V AC Network Unit Generation and Distribution
LNE	No-Break 220V AC Power Generation and Distribution
LOCA	Loss-of-Coolant Accident
LOPP	Life of Plant Plan
LTO	Long-Term Operation
LV	Low Voltage
MC	Metallic Components (ASME classification of subcomponents of the containment structure)
MEQ	Mechanical Equipment Qualification
MoA	Memorandum of Agreement
µSv/a	Microsievert per annum
mSv/a	Millisievert per annum
MSDF	Municipal Spatial Development Framework
MSL	Mean Sea Level
MSLB	Main Steam-Line Break
MV	Medium Voltage
MW	Megawatt
MWe	Megawatt electric
NCRWM	National Committee on Radioactive Waste Management
Necsa	South African Nuclear Energy Corporation
NEXCO	Nuclear Executive Committee
NIL	Nuclear Installation Licence
NNR	National Nuclear Regulator (the Regulator)
NOU	Nuclear Operating Unit
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRWDI	National Radioactive Waste Disposal Institute
NSA	Nuclear Safety Assurance

Abbreviation/ Trigram	Definition
NSC	Nuclear Safety Culture
NSRB	Nuclear Safety Review Board
NSRC	Nuclear Safety Review Committee
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OE	Operating Experience
OEM	Original Equipment Manufacturer
OIL	Operational Intervention Level
OSG	Original Steam Generator
OSGISF	Original Steam Generator Interim Storage Facility
OSP	Operational Safety-Related Programme
OTS	Operating Technical Specifications
Pa	Pascal
PAZ	Precautionary Action Planning Zone
PER	Pressure Equipment Regulations
PIE	Postulated Initiating Event
PM	Preventive Maintenance
POC	Programmes Oversight Committee
PSA	Probabilistic Safety Assessment
PSHA	Probabilistic Seismic Hazard Analysis
PSR	Periodic Safety Review
PTR	Spent Fuel Pit Cooling System
PVC	Poly-Vinyl Chloride
PWR	Pressurised Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
QA	Quality Assurance
RAR	Risk Assessment Report
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pressure System
RCV	Chemical and Volume Control System
RD	Requirements Document
RG	Regulatory Guide
RIS	Safety Injection System

Abbreviation/ Trigram	Definition
RLE	Review-Level Earthquake
RPC	Radiation Protection Certificate
RPE	Vent and Drain System
RPO	Radiation Protection Officer
RPR	Reactor Protection Logic System
RPV	Reactor Pressure Vessel
RPVH	Reactor Pressure Vessel Head
RPVI	Reactor Pressure Vessel Internal
RRA	Residual Heat Removal System
RRI	Nuclear Component Cooling System
RTP	Radioactive Waste (Radwaste) Tracking Programme
RVSP	Reactor Vessel Surveillance Programme
SALTO	Safety Aspects of Long-Term Operation
SAMG	Severe Accident Management Guideline
SAR	Safety Analysis Report
SBE	Equipment for Hot Maintenance Store and Laundries
SCEP	Safety Culture Enhancement Programme
SCO	Suitability for Continued Operation
SDRG	Safety Documentation Review Group
SEC	Essential Service Water System
SED	Demineralised Water Distribution System
SEK	Monitoring and Discharge of Conventional Island
SF	Safety Factor
SG	Steam Generator
SGR	Steam Generator Replacement
SOER	Significant Operating Event Report
SPLUMA	Spatial Planning and Land Use Management Act
SRA	Safety Reassessment
SRSM	Safety-Related Surveillance Manual
SSC	System, Structure, Component
SSE	Safe Shutdown Earthquake
SSHAC	Senior Seismic Hazard Analysis Committee
SSR	Site Safety Report

Abbreviation/ Trigram	Definition
SVA	Auxiliary Steam Distribution System
TEG	Gaseous Waste Treatment System
TEP	Boron Recycle System
TEU	Liquid Waste Treatment System
THA	Tsunami Hazard Assessment
TISF	Transient Interim Storage Facility
TLAA	Time-Limited Ageing Analyses
TLD	Thermoluminescent Dosemeter
TOMP	Technological Obsolescence Management Programme
TOP	Technological Obsolescence Programme
TPU	Thermal Power Uprate
TRS	Technical Requirement Specification
TSC	Technical Support Centre
UCP	Upper Core Plate
UIL	Unfiltered In Leakage
UNSCEAR	United Nations Special Committee on the Effects of Atomic Radiation
UPZ	Urgent Protective Action Planning Zone
WANO	World Association of Nuclear Operators
WBC	Whole-Body Counter
WENRA	Western European Nuclear Regulators' Association
XCA	Auxiliary Boiler System
XLPE	Cross-Linked Polyethylene

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[300] WASH-1250-1973: The Safety of Nuclear Power Reactors and Related Facilities

5.0 Long-term Operation of Nuclear Power Plants Requirements Framework

The section defines the main national regulations that contain the requirements for LTO, and main international safety standards used for the LTO assessment of the facility. The exhaustive list of regulations and standards applicable to the LTO assessments is listed in § 4.3 of the document.

5.1 National Requirements

5.1.1 Acts

- National Nuclear Regulator Act, NNR Act 47 of 1999
- Occupational Health and Safety Act, Act No.85 of 1993 and Regulations
- National Environmental Management Act, Act No. 107 of 1998
- Critical Infrastructure Protection Act 8 of 2019

5.1.2 Regulations

- Department of Mineral Resources and Energy, Regulations on the Long-Term Operation (LTO) of Nuclear Installations, Regulation, R.266
- Department of Mineral Resources and Energy, Regulations on Safety Standards and Regulatory Practices, R.388

5.1.3 Regulatory Documents (RDs) and Regulatory Guides (RGs)

- NNR RD-0014, Emergency Preparedness and Response Requirements for Nuclear Installations
- NNR RD-0022, Dose Limitations for Koeberg Nuclear Power Station
- NNR RD-0024, Requirements on Risk Assessment and Compliance with Principal Safety Criteria for Nuclear Installations
- NNR RG-0019, Interim Regulatory Guide: Interim Guidance on Safety Assessments of Nuclear Facilities
- NNR RG-0027, Ageing Management and Long-Term Operations of Nuclear Power Plants
- NNR RG-0028, Interim Regulatory Guide: Periodic Safety Review of Nuclear Power Plants

5.2 International Safety Standards

- IAEA, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, GSR Part 3

- IAEA, General Safety Regulations, Safety Assessment for Facilities and Activities, GSR Parts 4
- IAEA, General Safety Regulations, Preparedness and Response for a Nuclear or Radiological Emergency, GSR Part 7
- IAEA, Specific Safety Report, Safety of Nuclear Power Plants: Design, SSR 2/1
- IAEA, Specific Safety Report, Safety of Nuclear Power Plants: Commissioning and Operation, SSR 2/2
- IAEA, Specific Safety Report, Site Evaluation for Nuclear Installation, SSR 1
- IAEA, Specific Safety Guide, Ageing Management and Development of a Programme for Long-term Operation of Nuclear Power Plant, SSG-048
- IAEA, Specific Safety Guide, Periodic Review of Nuclear Power Plant, SSG-25

6.0 Scope of the LTO Assessments

Based on the regulatory LTO requirements, the section details the limitations related to the operation of the current facility. It provides the basis for the LTO assessments required for the justification of LTO and provides an overview of the LTO programme for the assessment activities.

6.1 Limitations Related to the Facility

6.1.1 Design Service Life Limitation

The Koeberg site consists of two pressurised water reactor (PWR) units constructed between 1976 and 1985. The first unit (Unit 1) was commissioned for commercial operation in 1984 and the second (Unit 2) in 1985. The safety analysis report for the facility indicates that the nuclear steam supply system (NSSS) has been designed to withstand all transients anticipated during the 40-year service life of each unit, assuming an 80% load factor. The 40-year service life ends in July 2024 and November 2025 for Unit 1 and Unit 2, respectively.

Design codes and standards have improved since the construction of Koeberg more than 40 years ago. Until the third PSR, the design of the facility had not been comprehensively reviewed against the current codes and standards, except during plant modifications, which are performed utilising applicable modern codes and standards. Therefore, to ensure safe LTO, there is a need to determine the safety of the facility and the extent to which the facility conforms to national and international standards and codes.

The Koeberg nuclear safety policy objectives include commitments to continuously improve the safety of the plant, and Eskom has adopted the IAEA safety reassessment process to routinely review the plant against a credible referential and not to rely only on regulatory requirements to drive plant safety improvements. Électricité de France (EDF) is often referred to as the "technical safety reference for Koeberg" due to the similarity in design to Koeberg, its rigorous safety improvement programme, and the close cooperation between the South African and French nuclear regulators and, as such, is used to form the basis for many of the technical decisions and the general operating rules (GORs) employed at Koeberg. Eskom is affiliated with nuclear agencies such as the Institute of Nuclear Power Operations (INPO), the World Association of Nuclear Operators (WANO), and EPRI, among others, and, thus, follows and implements WANO/INPO lessons learnt through formal WANO peer reviews, and assessing significant operating event reports (SOERs), etc. to improve the safety of the plant.

Therefore, plant changes have been performed throughout the life of the facility to address lessons learnt from operational experience and ensure that plant design remains current and in line with modern safety standards and in accordance with Eskom's asset management strategy.

6.1.2 Ageing Management Philosophies Improvements

Since the commissioning of the plant, ageing management philosophies to manage the ageing effects of SSCs have improved. Koeberg has been improving the ageing management programmes mainly as a result of operating experience. However, in order to ensure safe LTO, the current ageing management framework at Koeberg requires assessment to ensure that it is in line with international good practices applicable to the management of important safety SSCs.

6.1.3 Installation Licence Limitation

The nuclear installation licence (NIL-01 Variation 19), which stipulates conditions of operation for the facility, is valid until 21 July 2024, unless amended for subsequent licensing stages, including LTO, or varied, suspended, or revoked. Therefore, Eskom must obtain licence approval from the NNR to continue operations beyond the 40-year life. This safety case supports an additional 20 years, from 2024 to 2044 (Unit 1) and 2045 (Unit 2).

6.2 Requirements for the LTO Justification Safety Case

Generally, a safety case for LTO describes the nuclear and radiological risks associated with the operation of the facility beyond established time frames. The safety case describes the risk in terms of the hazards presented by the facility, the site, and the modes of operation, including potential faults or accidents, and the practicable measures that need to be implemented to prevent or minimise the harm. It demonstrates that provisions have been made to either eliminate or mitigate the risks such that the public and the environment will not be subjected to undue risk due to the operations. Safety assessments must be performed to identify the risks and determine the provisions required to eliminate or mitigate the risks.

In terms of the 'Regulations on the Long-Term Operation of Nuclear Installations' (Department of Mineral Resources and Energy (DMRE), R. 266) [240], any licensee wishing to operate a nuclear installation beyond an established time frame defined in the respective nuclear installation licence [286] shall lodge an application to operate the respective nuclear installation beyond an established time frame. Such application shall be supported by a safety case to demonstrate continued safe operation of the nuclear installation for the period of LTO, addressing requirements as stipulated in the LTO regulations [240]; that is, the safety case shall, among others:

- demonstrate compliance with relevant regulatory safety criteria and requirements;
- base the application on the results of a safety analysis, with due consideration of the ageing of SSCs and PSR;
- provide an overall assessment of the safety of the nuclear installation and justification for continued safe operation;
- demonstrate the availability of financial and human resources, as well as knowledge management; and
- include the necessary safety improvements in the application, including refurbishment, provision of additional structures, systems, and components, additional safety analyses, and engineering justifications, to ensure that the licensing basis remains valid during the LTO period.

Therefore, based on the requirements mentioned above, the scope of the LTO preparation activities included the following:

- Safety assessments specifically related to the management of ageing of the plant beyond the originally established time frame set by NIL-01 Variation 19
- PSR to determine the suitability of the plant for continued operation, including LTO
- Plant changes required to manage the effect of ageing of the plant and those changes needed to ensure that equipment qualification remains valid for the intended period of LTO
- Assessments related to the impact of LTO on other safety-related programmes associated with the protection of the public and the environment

The LTO assessments were performed using current national regulatory requirements and international standards and codes.

The programme for the LTO preparation activities was developed in accordance with the requirements of the regulatory guidance, which clearly states what should be included in the LTO safety case. Therefore, the scope of the safety case addresses all the requirements listed in the LTO regulations using the guidance in RG-0027 [294] and, hence, ensures that there is no undue risk to the workers, the public, or the environment.

The safety case sets expectations and guidance for the processes that will be used in the future if the hazards are to be adequately controlled.

This safety case aims to demonstrate that the design basis, the design of the plant, and the ageing management philosophy are adequate for continued operation. In addition, it demonstrates that adequate organisational provisions, skills and expertise, and management systems are in place to support safe LTO.

6.3 Overview of the LTO Assessments Programme

An LTO programme for the preparation activities was developed based on the licensing framework and the regulatory requirements. An overview of the LTO assessments programme based on the requirements of RG-0027 [294] is depicted in Figure 6-1. A detailed description of the scope of these assessments is given in § 7.0.

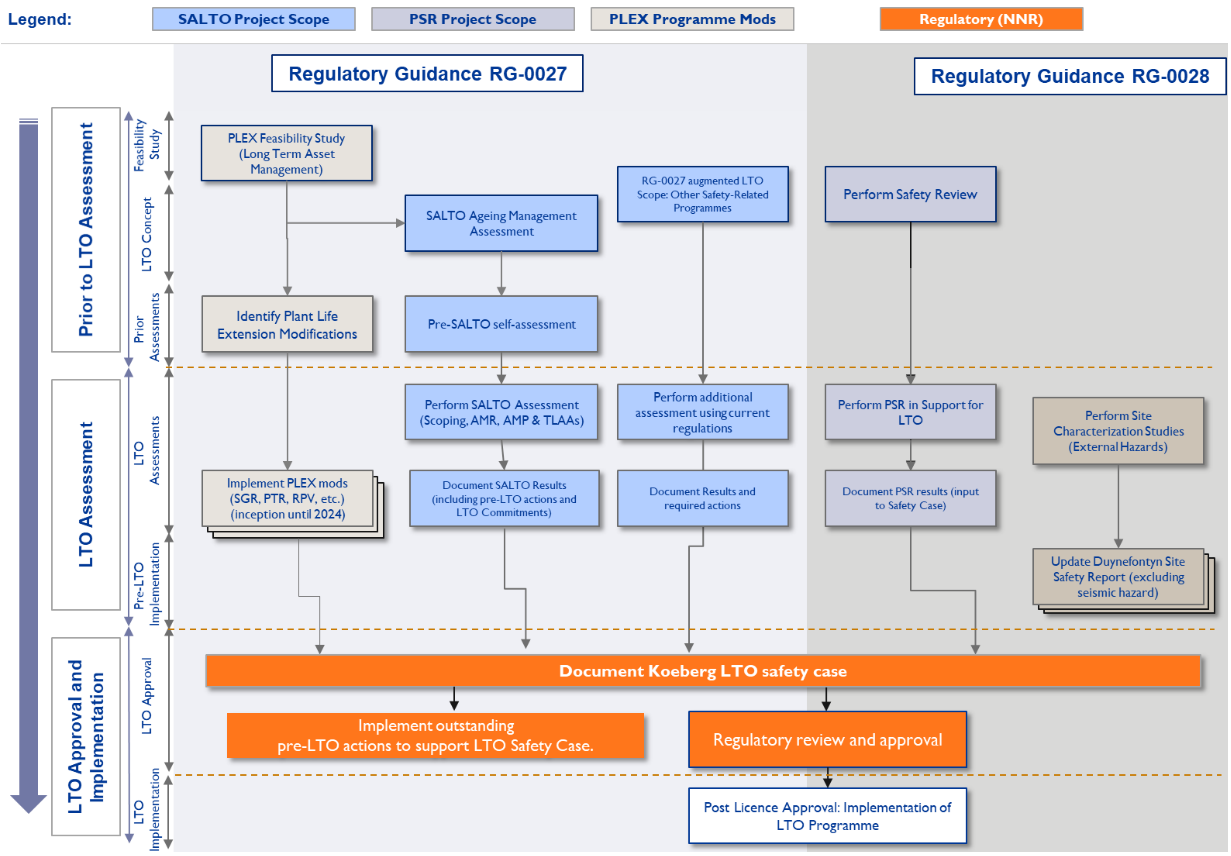

Figure 6-1: *LTO Assessment Activities*

7.0 Description of the LTO Assessment Activities

The purpose of the section is to further describe the LTO assessment activities introduced in § 6.3, to describe their objectives, and – where applicable – to discuss the integration of these activities.

Koeberg's LTO methodology is aligned with the requirements of RG-0027 [294], except for the PSR, which was conducted in the assessment phase instead of the pre-LTO assessment phase; refer to Figure 7-1 for the adaptation. The adaptation of the methodology was accepted by the NNR.

The ageing management assessments commenced prior to the PSR. (The result could be that the gaps identified in the PSR relating to ageing management were already in the process of being addressed in the ageing management assessments, which might have had an impact on the safety significance of those gaps.) Where the impact of a particular gap (also referred to as a deviation) is important for the justification of safe LTO, the details are discussed in the relevant subsections of § 9.0.

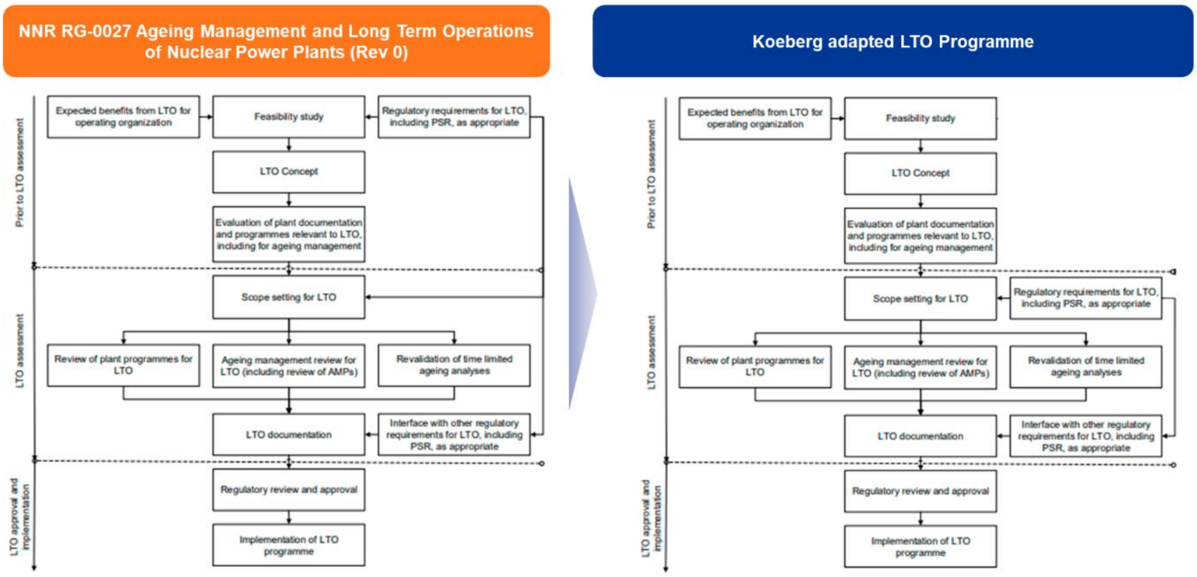

Figure 7-1: *Comparison Between RG-0027 LTO Assessments and Koeberg-adapted LTO Assessments*

7.1 Plant Life Extension Feasibility Studies

The feasibility study was undertaken to address strategic elements, such as the need for electrical power, an economic assessment, and issues concerning diversity in supply, considering that nuclear safety took precedence over electricity production. The feasibility study followed an internationally accepted approach and methodology documented in the IAEA-TECDOC-1309 (*Cost Drivers for the Assessment of Nuclear Power Plant Life Extension*) [269]. The objective of the feasibility study was to determine which plant modifications were required to safely and reliably extend the life of Koeberg and for continued operation to remain viable. The feasibility study is detailed in K08016VAR (*Koeberg Plant Life Extension*) [133], and the outcomes of the study are discussed in § 9.4.1.

7.2 Periodic Safety Review

A PSR is a comprehensive review of all nuclear power plant (NPP) aspects important to safety, carried out at regular intervals, and is used in support of the decision-making process for long-term operation. It provides an effective means to obtain an overall view of actual plant safety and the quality of the safety documentation and to determine reasonable and practical safety improvements needed for an acceptably high level of safety.

In accordance with IAEA SSG-25 (*Periodic Safety Review for Nuclear Power Plants*) [264], the objective of a PSR is to determine the following by means of a comprehensive assessment:

- The adequacy and effectiveness of the arrangements and the SSCs that are in place to ensure plant safety until the end of a planned operation
- The extent to which the plant conforms to current national and/or international safety standards and operating practices
- Safety improvements and timescales for their implementation
- The extent to which the safety documentation, including the licensing basis, remains valid

The third PSR for the facility commenced in August 2019 and was concluded in June 2022, and its outcomes are documented in 331-607 (*KNPS 3rd Periodic Safety Review Final Report*) [114]. It was conducted in accordance with RG-0028 [295] to determine compliance with national safety criteria and requirements and to determine the extent to which the plant met the PSR objectives mentioned above. It was used to support the additional 20 years of intended LTO by determining the suitability of the facility for continued safe LTO. The safety aspects mentioned below were considered important for LTO:

- Plant design
- The actual condition of SSCs important to safety
- Equipment qualification
- Ageing
- Deterministic analysis, specifically analysis involving time-limiting assumptions relating to the proposed lifetime
- Programmes for promoting a safety culture focused on the pursuit of excellence in all aspects of safety management and human factors
- The process of ensuring that key technical competencies would be sufficient for future operation
- A management system that addressed quality and configuration management

The review was conducted in four main phases, namely:

- the preparation phase, in which the scope, review requirements (that is, review regulations, codes, and standards), methodology, and acceptance criteria were agreed with the NNR;
- the safety factor (aspects) review phase, in which all the safety aspects were assessed against the review requirements to determine strengths and deviations;
- global assessment, which considered the cumulative impact of all the findings, that is, the strengths and deviations, and proposed safety improvements to address all the deviations and global issues in a PSR integrated implementation plan (IIP); and
- final reporting, which was a summary of all the activities and the results.

A graphic representation of the periodic safety review as it has been applied in support of LTO is depicted in Figure 7-2.

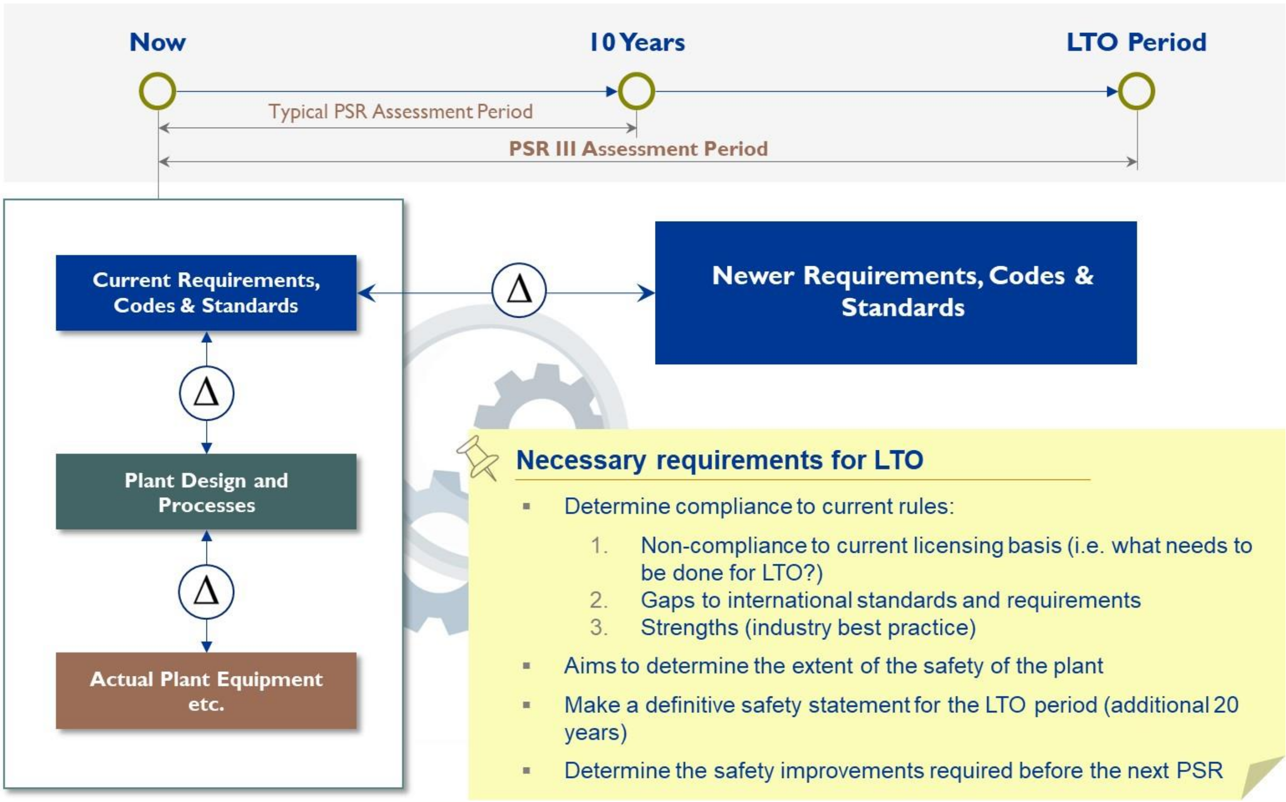

Figure 7-2: *Periodic Safety Review Methodology*

Where deviations would have an impact on the requirements for safe entry into LTO – which might require further assessments – these assessments were performed to determine the safety improvements necessary to meet the LTO requirements. Among others, this was the case for the ageing review in the PSR. To meet the LTO requirements, the ageing assessments were required prior to entry into LTO, and these are discussed in § 7.3.

7.3 SALTO Ageing Management Assessments

Ageing management is implemented to ensure that the effects of ageing will not prevent SSCs from being able to fulfil their required safety functions throughout the lifetime of the facility. It takes account of changes that occur with time and use [265]. This requires addressing the effects of both the physical ageing of SSCs, resulting in the degradation of their performance characteristics, and the non-physical ageing (obsolescence – SSCs that are no longer manufactured or supported) of SSCs [265].

The ageing management assessments discussed in this report exclude ageing management for the decommissioning phase of the facility. Ageing management related to decommissioning is addressed in the decommissioning strategy and plan. These documents have been submitted to the NNR.

Therefore, the objective of the ageing management assessments was to evaluate the technical systems to ensure that the required safety margins were maintained. In the assessments, the reliability of SSCs was evaluated, considering the possible time-dependent degradation. The ageing management assessments were done in compliance with IAEA safety standards and their consistency was reviewed by the IAEA SALTO peer review service. In accordance with SSG-48 (*Ageing Management and Development of a Programme for Long-Term Operation of Nuclear Power Plants*) [265], the approach to ageing management assessments was as given in the main steps below.

1. Organisational arrangements

   To ensure that ageing management has been included in the policy and objectives of the facility ensure that suitable organisational and functional arrangements have been established in which all necessary members of staff (internal and external) involved in and/or supporting ageing management are assigned responsibilities for ageing management.

2. Data collection and record keeping

   This is to ensure that a data collection and record-keeping system as a necessary base for the support of ageing management is in place to facilitate obtaining the necessary quality and quantity of ageing-related data from plant operation, maintenance, and engineering, including documentation from suppliers.

3. Scope setting

   It is a systematic process to identify SSCs important to safety subject to ageing management.

4. Ageing management review (AMR)

   AMR is the systematic process of assessing ageing effects and the related degradation mechanisms that have been experienced or are anticipated, including the evaluation of the impact of these on the capability of the important-for-safety SSCs to perform their intended functions. Included in the AMR are the following aspects or elements:

a) Time-limited ageing analyses (TLAAs)

Included in the ageing management review are the TLAAs associated with SSCs that are important to safety.

b) Ageing management programmes (AMPs)

This element involves the identification of AMPs to manage the ageing effects and degradation mechanisms. This step is performed to ensure that the AMPs are consistent with the attributes of an effective ageing management programme.

5. Documentation of ageing management

This step is aimed at ensuring that the assumptions, activities, evaluations, assessments, and results of the evaluation of the plant programme for ageing management are documented in accordance with national regulatory requirements for the facility; this must be documented in accordance with RG-0027 [294].

The justification for safe LTO utilises the results of the ageing management assessments to demonstrate the adequacy and effectiveness of the ageing management programmes to provide confidence that the ageing of the plant will be managed in line with national regulations and international standards and practices during the period of LTO. The ageing management assessments for LTO were performed in accordance with the requirements of the regulatory guidance RG-0027 [294].

A summary of the ageing assessment process followed for the facility is depicted in Figure 7-3 below.

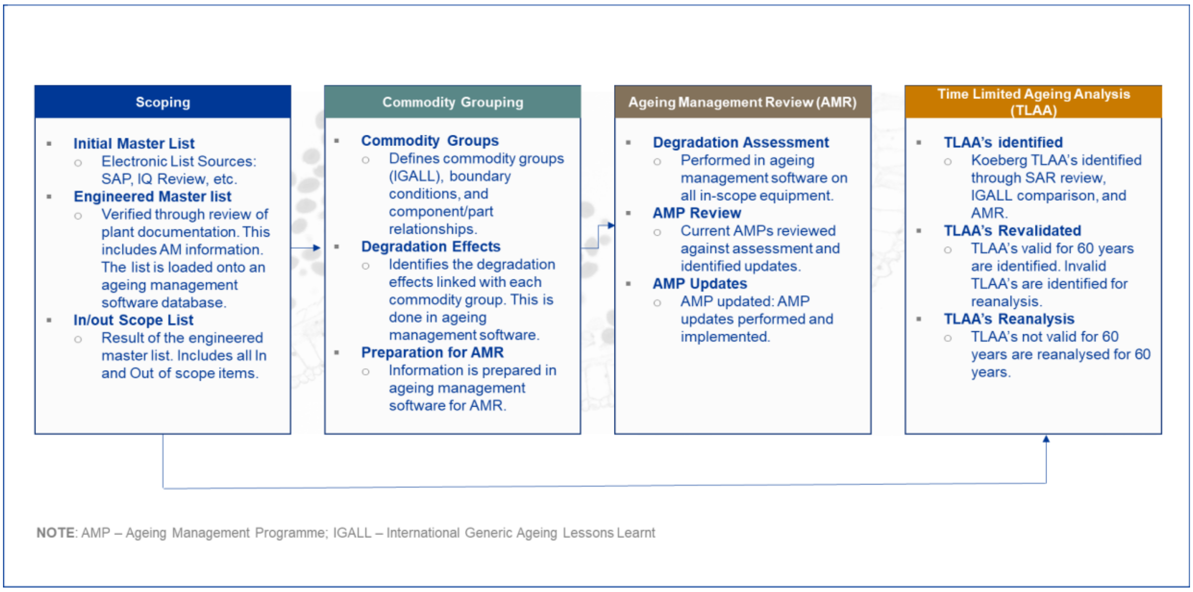

Figure 7-3: *Ageing Management Assessment Process*

7.4 Operational Safety-related Programmes (OSPs)

RG-0027 [294] requires that the impact of LTO on safety-related programmes other than those essential for ageing be assessed. The other safety-related programmes include the following:

- Nuclear security
- Radioactive waste management, including adequate safe storage of spent nuclear fuel
- The environment
- Radiation protection
- Emergency planning

Included in the assessment of the impact of LTO on these programmes is the assessment of the ageing management of the SSCs or equipment used in support of the effectiveness of these programmes. These SSCs may include those not deemed important to safety according to criteria used in the relevant SSC classification safety standard or regulatory guidance documents. The objective of the assessment is to demonstrate that SSCs needed to support these licence-binding programmes are adequately managed throughout the intended period of operation.

7.5 Site-specific Characterisation

The Koeberg Site Safety Report (KSSR) contains information relating to the hazards applicable to the facility. During the second periodic safety review of the plant in 2008, it was found that the KSSR was outdated; that is, the information relating to the hazards was no longer accurate, or new hazards needed to be considered in the site characterisation, and therefore, the KSSR had to be updated with the latest information relating to site-specific hazards. Any changes to the site characterisation might result in plant design basis or design changes.

Eskom has adopted the convention of referring to the site as “Duynefontyn” instead of “Koeberg”, which was used during the previous revision of the site safety report. This is in recognition of the potential to locate an additional nuclear installation(s) at the site. The title of the Koeberg Site Safety Report was, therefore, changed to the “Duynefontyn Site Safety Report” (DSSR) for alignment with the official deeds-registered name of the greater portion of the Eskom property, namely, the farm Duynefontyn No. 1552.

The justification for safe LTO, as detailed in RG-0027 [294], should include a description of any design basis reassessment. To determine any need for design-based reassessment, studies relating to the update of the site characterisation had to be performed as part of the LTO assessments.

Apart from providing information on site characteristics related to the current facility, the DSSR evaluates and demonstrates the suitability of the Duynefontyn site for accommodating additional new nuclear installation(s). This allows for assessing the cumulative radiological impact of the site on the public and the environment.

8.0 LTO Safety Case Methodology

This section outlines the methodology applied in developing the safety case for LTO. The LTO assessments have interfaces and are, therefore, interlinked. The LTO assessments were performed on various projects in line with the discussion in § 7.0. Thus, the relevant outputs from the LTO assessments are referenced in the justification for LTO as a source of evidence. These deliverables contain, among others, the gaps identified during the assessments, the actions required to resolve these gaps, and the time frame for the resolution of these actions. The actions are categorised into two categories in this safety case, namely, actions required to be completed prior to entry into LTO and actions to be completed during the LTO period to ensure safe LTO.

8.1 Structure and Content of the Safety Case

The NNR approved the LTO safety case structure and content document [58], and this document was used as the basis for the scope of topics to be addressed in the safety case. The structure and content of the safety case document were developed to meet the requirements of the interim guidance on LTO and ageing management of nuclear power plants [294].

Not all the actions to support the arguments of the safety case will have been completed when the safety case is submitted. As a result, the approved safety case content includes an LTO IIP (list of commitments), that is, all actions to be resolved before entry into LTO to ensure that the safety case remains valid.

Except for the outstanding TLAAs, all other LTO assessments have identified the actions that are required prior to entry into LTO. In accordance with RG-0027 [294], ageing management actions required to demonstrate safe LTO are provided in the safety case.

8.2 Compilation Methodology

The claim-argument-evidence approach was adopted in developing the safety case. Based on the conclusions of the LTO assessments, claims or assertions are made, and arguments are provided to support the claims. The evidence for the arguments is provided by the LTO assessment documentation. Thus, § 9.0 of the safety case provides the justification for safe LTO based on the adopted approach, while § 10.0 provides a summary of the overall safety of the plant and its readiness to enter LTO.

8.3 Use of PSR Deviations in the Safety Case

Not all the deviations raised during the review phase of the PSR were included in the safety case. Only deviations relevant to the topics being discussed are highlighted in the safety case. This is not deemed a concern because all the deviations raised in the PSR were assessed in the global assessment and are considered fully in the suitability of the plant for LTO.

8.4 Quality Aspects of the Safety Case

The LTO assessments completed in preparation for LTO and used in this safety case were performed by a combination of local and international experts. Industry experts were utilised to complement the skills and expertise requirements for the LTO assessments.

To ensure that the PSR was of the highest quality, the IAEA provided training for the lead internal personnel and external organisations. All the PSR activities were subject to technical and independent reviews. Koeberg also obtained support from the IAEA to review the PSR basis document, the safety factor (SF) requirements and review methodology documents, the global assessment (GA) and IIP methodology document, the SF reports, and the GA and IIP report.

The ageing management assessment utilised the 'IAEA Safety Report Series No. 82, Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL)' [256] as the main source of operating experience, and the assessments were performed by the original equipment manufacturer. Additionally, two SALTO peer missions and a SALTO technical mission were conducted to support the performance of the ageing assessments. This approach ensured that national and international requirements for ageing assessments were satisfied.

The DSSR studies were performed by specialist organisations.

In accordance with the integrated management systems of the facility, the LTO assessments followed rigorous approval processes.

Furthermore, to ensure the quality of the justification for safe LTO, the following measures were taken:

- The local safety case compilation team was supplemented by international experts on LTO.
- The safety case was independently reviewed by local and international industry experts.
- The safety case went through a rigorous approval process, which included both technical and safety forums.

9.0 Justification for Safe LTO

This section provides the justification for safe LTO utilising the safety case methodology discussed in § 8.0. The justification was based on the outcomes of the LTO assessments discussed in § 9.1 to § 9.9.

9.1 Summary of the PSR

This section provides a summary of the PSR by providing the results of the safety factor reviews and the global assessment (specifically focusing on the PSR aspects linked to LTO in accordance with § 7.2).

The scope of the PSR is contained in 240-134382460 (*PSR basis document – 3rd Periodic Safety Review for Koeberg Power Station*) [44], which was approved by the NNR. The review was performed utilising a combination of the national requirements, and IAEA and Western European Nuclear Regulators' Association (WENRA) requirements. The PSR was conducted based on the requirements of the interim regulatory guidance RG-0028 [295]. Detailed outcomes of the PSR are discussed in 331-608 (*KNPS 3rd Periodic Safety Review Global Assessment and Integrated Implementation Plan Report*) [115] and 331-607 (KNPS 3rd *Periodic Safety Review Final Report)* [114].

9.1.1 Outcomes of the Safety Factor Reviews

Approximately 1 150 requirements were assessed across all the PSR safety topics, and as a result, 113 deviations were raised. The deviations were graded commensurate with their safety significance based on the risk grading method contained in the PSR basis document. Of the 113 deviations, a single deviation related to the control room envelope (CRE)that did not meet the applicable dose criterion during accidents was graded as a high. To address all the deviations raised, 93 safety improvements were proposed. (Details of the safety improvements are discussed in § 9.1.3.) The review was conducted to assess the suitability for continued safe operation for both the period until the next PSR and the end of extended life, that is, LTO.

9.1.1.1 Plant Design Review Results

A total of 153 consolidated requirements were reviewed. This review resulted in the identification of three strengths, 32 observations, and 15 deviations. The review included the assessment of corrective actions that had been identified in historical safety reassessments (SRA) as well as the adequacy of the plant design for LTO.

The evaluation of the recorded deviations concluded that the deviations did not preclude safe plant operation or safe LTO and had to be corrected in accordance with the ranking and scheduling applied to the safety improvements.

The assessed plant design processes and procedures were adequately robust to maintain the ongoing integrity of the plant design and safety case. This conclusion was deemed valid for all plant states throughout the forward-looking PSR period and into LTO. If any unforeseen issues involving plant design were to occur going into LTO, it was judged that the existing processes and arrangements in place with regard to plant design aspects were comprehensive and robust enough to ensure that the issues were managed effectively.

It was identified that the completion and close-out of modifications originating from historical SRAs had been a challenge over the review period of the third PSR. This was evidenced by the inability to close out the second SRA (*SRA-II*) identified corrective actions and slow implementation of the external event review initiative (EERI) programme's planned initiatives. The above may potentially be indicative of affordability, staff resourcing issues, contractor support, or the redeployment of key resources to “higher”-priority work before project finalisation. These may also represent a nuclear safety culture concern, which will be further challenged, as the implementation of the PSR safety improvements is coincident with the implementation of steam generator replacement (SGR) and other improvements to support LTO.

It should be noted that similar issues were identified in other safety factor reviews and were addressed in the global assessment. In addition, it is acknowledged that the level of robustness demonstrated throughout this PSR represents a significant step change compared to previous SRAs. This is evident from the inclusion of the IIP from the outset of the project and the KOU management commitments being made on the execution of the plan. In the context of the plant design review, it is recommended that the design-related corrective actions identified from historical SRAs be implemented or suitable and sufficient arguments for dispositioning the issues raised be adequately documented.

In conclusion, the plant design review met the objective of plant design as required in RG-0028 [295]. Despite the deviations identified, the overall evaluation of the plant design review outcomes concluded that the plant design, including documentation and design processes, was adequate when reviewed against the current licensing basis, and national and international standards, requirements, and practices. The deviations raised need to be resolved; however, they do not preclude current safe plant operation and safe LTO.

Specific outcomes of note relating to LTO

The evaluation of TLAAs identified 111 TLAAs, of which 11 were reported as requiring validation for LTO at the time of completing PSR (Of these, 105 have since been validated, with six outstanding). It is noted that the equipment qualification and ageing review evaluation reports had identified the outstanding TLAAs required for LTO that were currently in the process of being revalidated or validated for the newly identified TLAAs. Therefore, no new deviations were raised in the plant design review for the same issue. Ageing management was addressed specifically in the ageing review, and the SALTO ageing management assessment scope includes reviewing all age-related design inputs and the consideration of time-limiting design concerns.

There is a full EERI programme of works currently being rolled out to address the post-Fukushima deficiencies (including plant modifications) that were identified to benefit from the Fukushima lessons learnt. Overall, the station has taken significant steps in the development and implementation of key work programmes following on from significant internal and external operating experience (OE) since the last PSR and in providing assurance for safe LTO.

The assessment undertaken in relation to the suitability of materials selected for SSCs important to safety largely focused on changes to design codes and standards associated with the three principal barriers to the release of radioactive material:

- Barrier 1 – fuel design

- Barrier 2 – primary circuit design
- Barrier 3 – containment design

A single deviation was identified against requirements relating to multiple codes and standards-related findings for the mechanical design aspects of the plant. As a result, a deviation was raised in this regard. It will enable further assessments to be undertaken on the individual findings relating to the mechanical aspects of the codes and standards review, in a proportionate and graded approach. Given the uncertainty to the extent that the findings have been addressed by ongoing programmes of work to support LTO (for example, the SGR project), considering the information available during the plant design review, the total risk posed associated with this deviation was graded as “low”.

Overall, the plant design review assessment demonstrated that procedures are in place to fulfil the requirements of a design management system and configuration management system. The procedures are broadly aligned with international good practices and ensure that plant design changes are fit for purpose and controlled in line with the configuration management process. No issues were identified with regard to configuration management that presented a risk that could prevent the facility from entering LTO.

9.1.1.2 Actual Condition of SSCs Important to Safety

Primarily, the review of the actual condition of SSCs important to safety was aimed at demonstrating compliance with the current licensing basis and benchmarking station practices against the latest standards and international guidelines with regard to all plant activities involving the ageing management of SSCs and the revalidation of the actual condition of SSCs important to safety in order to assess their performance and reliability during operational life for the entire period of LTO.

The following deviations raised by the review of the actual condition of SSCs important to safety were relevant when considering the impact on continued safe operation into LTO:

- Civil structures

  Based on the results obtained in the SSC health evaluation, it was concluded that the condition of safety-related civil structures might potentially affect nuclear safety in future if there were to be significant delays in the execution of repairs, causing the condition to deteriorate. Although civil monitoring remained good, and the civil structures remained intact, better management focus and preventive maintenance planning were required to avoid further deterioration of civil structures. Station management was aware of issues related to examinations, maintenance, inspections, and tests that might affect plant safety; however, in the case of civil structures, a certain lack of management focus with regard to the slow repairs remained a concern.

  Three deviations were raised related to the integrity of safety-related civil structures, which could be compromised if repairs and protection measures are not timeously implemented, improved health reporting for the civil structures, and management focus on the execution of civil repairs. Timely resolution of these deviations will ensure continued safe operation into LTO.

- Plant programmes for ageing management

  The review recognised that the AMPs were in the process of being fully aligned with international good practices; however, all items important for nuclear safety and all internationally known significant safety issues had been considered. Several mitigating actions and restorative activities remained ongoing as part of the ageing management process. This is in line with the findings of the ageing review, which are discussed in the ageing outcomes.

  The review focused on the mature ageing-related plant programmes such as the maintenance programmes, safety-related surveillance programme, in-service inspection programme, in-service testing programme, etc. and concluded that these were comprehensive and had been implemented well, which ensured that the required safety functions of SSCs important to safety were fulfilled and would continue to be fulfilled over the LTO period.

- Obsolescence

  In general, no “high” or “medium” safety significance plant and equipment health issues or adverse impact on SSCs important to safety had been reported during the review, except for a few issues highlighted, including the negative trend observed regarding the health of the personnel airlock and equipment hatch (EPP) airlock penetrations. Airlock failures during local leak rate tests occurred regularly, and obsolescence issues had not been addressed proactively and with urgency. While the airlocks were currently fully operable, a deviation was raised to evaluate the safety risk associated with frequent failure of the airlock tests. Obsolescence remained a potential risk to plant reliability and availability of SSCs important to safety. This is in line with a deviation identified in the ageing review related to the incomplete technological obsolescence programme (TOP).

Overall, the review of the actual condition of SSCs important to safety concluded that all programmes associated with maintaining the condition of SSCs are adequate, have been implemented well, and provide confidence in the delivery of safety functions of SSCs important to safety until the next PSR and during LTO, with no significant impact on nuclear safety.

Provided the necessary safety improvement actions are carried out and continuous maintenance, monitoring, surveillance, inspections, and tests are performed effectively, the projected condition of SSCs important to safety is expected to remain acceptable for the next 10 years and for the entire period of LTO [114].

9.1.1.3 Equipment Qualification Results

The review found that the programmatic aspects of the EQ programme are met and that the EQ programme is well integrated into the station’s processes to ensure the ongoing qualification of qualified equipment into the period of LTO. It was also found that the preservation requirements, implemented in order to ensure that qualified equipment was maintained to ensure its qualification, are adequate. The report concluded that the EQ programme processes and procedures are robust and are continuously reviewed to ensure that equipment important to safety was qualified for the duration of LTO.

Out of the four deviations raised for the equipment qualification review, a single deviation related to the TLAAs is specifically related to the period of LTO. The deviation relates to the incomplete documentation for the revalidation of EQ TLAAs for LTO. The proposed treatment for this deviation states that “KNPS should complete the TLAAs for qualified equipment prior to LTO”.

A complete list of the EQ items requiring replacement and reanalysis, including the LTO integrated preparation plan (IPP), is provided in the EQ TLAA strategy. The list of EQ TLAAs requiring reanalysis of the qualified life is found in document L1124-GN-RPT-018 (*Time-Limited Ageing Analysis Based on Initial Environmental Qualification*) [214], and this document is referenced in 240-156945472 (*SALTO Ageing Management Assessment Report (Interim)*) [57]. The strategy above, considering the timelines, is implementable and achievable. It was furthermore found that many of the TLAAs required for LTO had been updated as part of the ongoing SGR project and justified a 60-year plant life.

A deviation was also raised relating to inadequate monitoring and trending of the actual environmental conditions affecting the ageing analysis of qualified equipment. Environmental parameters and service conditions applicable to design extension conditions (DECs) had not been derived and incorporated into the EQ programme, and seismic event intensities associated with DECs had not been considered in the EQ programme.

The identified observations relate to opportunities for enhancements, for instance, corrections to procedures reviewed during the assessment.

In conclusion, the review of this safety factor found that the requirements for EQ, derived from available international standards and industry good practice, are largely met. The four deviations that were identified are all graded as “low” and do not pose a significant concern from a nuclear safety perspective. The EQ programme ensured that qualified equipment important to safety is capable of performing its safety function when required.

The EQ programme adequately addresses the requirements identified in PSR to ensure ongoing qualification of qualified equipment for the current operating period and LTO [114].

9.1.1.4 Ageing Management Review Results

The scope of the review covered ageing management for the duration of LTO. The review concluded the following:

- The specific time for LTO had been considered and documented in the AMP documentation.

- The safety analysis report (SAR) did not currently contain any references to ageing management or LTO, which were required by RG-0027 [294]; however, there were corrective actions in place to complete the SAR update.
- The ageing management assessments conducted by the SALTO project resulted in several corrective actions to enhance the current plant documentation to support LTO.

A deviation with a “low” safety significance was raised for the ageing management programme not being comprehensive. The safety improvement to update the ageing management programme is contained in the PSR IIP [114].

The review identified that the revalidation of some TLAAs required for LTO was in progress. The impact of the outstanding revalidation is addressed in the SALTO ageing assessments, and these TLAAs will be revalidated prior to entry into LTO.

Regarding ageing management, the organisation and management system review assessed the adequacy of the organisational arrangements, and the human factors review assessed the adequacy of the resources to support LTO. No negative findings were identified.

The reviewed ageing management organisational aspects, the objectives, and the intended period of LTO were found to be well documented and met the requirements of the national regulations and international safety standards. The previous ageing management concerns raised under SRA-II had been actioned and effectively addressed.

The requirements dealing with the programmes, processes, and management methods to effectively manage ageing through LTO were largely satisfied. No “high” and “medium” nuclear safety significant deviations were identified.

The review of the requirements addressing activities required to identify, detect, and recognise degradation mechanisms and ageing effects identified gaps related to the end of the current life of the plant, and if the implementation of safety improvements were to continue as scheduled, there would be no barrier that would prevent safe entry into LTO.

A “low” safety significance deviation was raised related to the technological obsolescence programme. The deviation was to address the deficiencies in the proactive aspects of the programme. The review confirmed that safe LTO could be achieved safely if the current AMP strategy and safety improvements were to be executed and any obsolescence events treated proactively [114].

9.1.1.5 Deterministic Safety Analysis (DSA) Review Results

The DSA review was performed against the latest requirements described in IAEA, WENRA, and NNR documentation. Eskom’s approach to DSA was considered to be well managed and captured within the documentation and management systems. Some deviations were identified through the review, but it was judged that these deviations would not adversely affect LTO. Proposed safety improvement actions were recommended in order to close out these deviations.

For the DSAs that had not been reperformed in the frame of the SGR and were contained in the SAR, a lack of QMS records was identified. Considering the high level of quality requirements applied to the DSA analyses, it was, however, not expected that significant deviations in DSA quality could arise (because the DSA had largely been reperformed during the SGR) that would impair the safety of continued operation until the next PSR or during LTO. The associated deviation was, therefore, considered to be a “drop” in terms of safety significance, and it related to the incorrect positioning of the fuel assembly during core reloading analysis.

Overall, the review found that the requirements for LTO were sufficiently addressed and that the deviations identified would not adversely affect LTO. It was concluded that Eskom’s approach to the DSA is well-managed and captured within the documentation and management systems. A total of 19 deviations were identified through the assessment, but these deviations do not adversely affect continued operation or LTO. Proposed safety improvement actions were identified in order to close out the deviations to ensure that the facility improved alignment with existing requirements and international references [114].

9.1.1.6 Organisation, Management Systems, and Safety Culture Review Results

The objective of the review was to confirm that the NOU organisation and management systems were adequate and appropriately documented to ensure the continued safe operation of the plant, both for current operation and the duration of LTO. In addition, the review confirmed that the NOU had an appropriate safety culture and evaluation method to monitor it.

The review of workforce planning established that the resources necessary to implement Eskom’s nuclear management policy were determined using a workforce plan, which provided the workforce requirements for the NOU for the period of 10 years from 2020 to 2030. The workforce plan is reviewed annually for the rolling period. This document (i.e., the workforce plan) outlines the strategic alignment of human resources with the business direction of the organisation, analyses the current workforce, determines future workforce needs, identifies the gap between the present and the future and implements solutions and/or strategies. The development of the workforce plan is a collaborative effort between the Human Resources Department and all the departmental business units within the NOU. The NOU has a documented long-term staffing strategy, implemented through the process of workforce planning.

The review concluded that the NOU has developed a workforce plan to determine current and future staffing requirements. The plan had been developed using international benchmarks. The workforce planning process ensures continuous review of resource requirements during the period of LTO. In its current state, the organisation and its associated processes remain adequate for LTO.

The documentation and implementation of the safety policy had been established; and managers communicate and reinforced the elements of the safety policy to all internal and external staff, as required. Managers encourage an open reporting culture and carry out plant observations, and management inspections are performed routinely. Arrangements are in place in the management system on how decisions affecting safety matters are raised, graded, and investigated.

An integrated management system (IMS) has been developed, implemented, maintained, reviewed, and monitored. There is management commitment to establishing, maintaining, and improving the management system, and accountability for the management system is documented. Roles and responsibilities for the safe operation of the plant are documented in the management system.

The high-level organisational structure and mandates are reflected in controlled documents. Responsibility for the safe operation and maintenance of the plant is clear. Changes to the organisation are managed through documented processes, and the requirement to inform the Regulator has been met. The current process did not require an assessment of the post-implementation of change, as recommended by international practices, and as a result, a deviation was raised. However, the safety significance of this deviation was assessed to be “low”.

Organisational processes had been established and documented. The documentation and records management (DRM) system has been established and implemented. Internal audits had identified anomalies with administrative controls of some of the processes reviewed. All audit findings were registered on DevonWay, and quality assurance processes were used to monitor implementation and closure. The archive storage capacity and the use of modern technology were a concern. The concern was mitigated by the use of Metrofile, which acts as a depository for records, and the information is backed up on the Eskom server every 24 hours.

The framework and strategy for communications with interested parties had been established, and the liaison with the Regulator and other authorities is well documented in the management system.

The procurement and supply chain management process were assessed against IAEA and NNR requirements. No deviations were raised in the review. The process as currently documented does not pose any risk to the LTO of the plant.

The safety culture enhancement programme exists and meets the regulatory requirements. Safety culture training took place during the nuclear safety awareness sessions and induction training. The assessment of the safety culture revealed that the management system requirements had been documented. Safety culture surveys were scheduled and undertaken. Safety culture self-assessments were also carried out.

Overall, the review concluded that the NOU organisation and management systems are adequate and appropriately documented to ensure continued safe operation of the plant, both currently and for the duration of the LTO and that the NOU has the appropriate safety culture and appropriate evaluation methods to monitor it.

9.1.1.7 Human Factors Review

The human factors review focused particularly on the strategic workforce plan and the sustainability of competent key technical staff, managers, and contractors required for the end of the present life and the duration of LTO. This included the review of the adequacy of staffing levels for positions such as maintenance, engineering, and radiation protection core critical skills, reactor operators, and emergency plan staff during normal, abnormal, and emergency conditions and for the duration of LTO.

Regarding staffing levels, the review concluded that the strategic workforce planning processes and procedures provide a framework for the development of strategic workforce plans in line with international and national good practices now and into the LTO period. In addition, in accordance with NIL-01, the human resources required to ensure the safe operation of the plant are demonstrated to the NNR on an annual basis.

The processes to recruit, retain, develop, and train staff are in place and in line with good practices to enable sufficient qualified staff for now and during LTO, provided the organisation continue to maintain these processes adequately.

The existing procedures and processes, including recruitment, selection, training, and assessments (including a process of authorisations), indicated that the competence requirements for individuals in key departments are currently adequate. Provided the organisation continue to maintain these requirements and benchmark as necessary, the documented requirements support safe plant operations into the LTO period.

In summary, the review concluded the following:

- The general staff selection and development procedures are comprehensive and well-aligned with international and national references. They describe, in detail, the full recruitment and selection value chain, the roles and responsibilities, and the variety of selection methods used.
- The review of procedures for licensed operators shows that these are detailed, comprehensive, and well aligned with international and national good practices and regulations.
- Overall, the organisation met the requirement that personnel assigned to perform emergency response functions have to be suitably competent, and overall staffing of this emergency response organisation is adequate, except in situations where emergency situations arose on both units simultaneously (A deviation was raised).
- The workforce plan procedure provides a framework for the development of strategic workforce plans in line with international and national good practices. The organisation demonstrated to the NNR on an annual basis that the required human resources were available to ensure the safe operation of the plant.
- For some departments, no evidence of documenting the safety impact caused by organisational changes prior to implementation of changes could be obtained (A deviation was raised).

- Koeberg had implemented a suite of training procedures and guidelines that ensured that the development and maintenance of the training programme were comprehensive. These procedures and programme guides were, in general, found to adequately address all training requirements. Exceptions to this included inadequate specification of training and periodic retraining requirements of operational support centre damage controllers and supervisors on the use of non-permanent equipment in applicable procedures, used when responding to accidents more severe than design basis accidents (a deviation was raised), and inadequate training on the IMS included in the leadership training programme (a deviation was raised).
- The facility was found to be compliant with all consolidated requirements with respect to training facilities. The status of training facilities is aligned with national and international good practices and supports nuclear safety.
- The review considered human factors engineering (HFE) and focused on the following in operations and maintenance:
  - Control room design
  - Emergency facilities
  - Modifications and design changes

  No deviations were identified where Koeberg did not meet the associated requirements. However, it was considered that HFE in the evaluation of modifications and design changes could be strengthened by making use of up-to-date guidelines on the implementation of HFE. An observation, as well as associated recommendations, in the form of a condition report was raised.
- The organisation adequately meets all requirements related to ongoing medical surveillance, which is well aligned with international and national good practices and regulations. A comprehensive suite of procedures exists, and the procedures are rigorously followed. According to such procedures, the organisation is currently well equipped with respect to the fitness for duty of all personnel and, particularly, a more rigorous process for the reactor operators.

Overall, the review concluded that human factors are well managed and documented in terms of station processes, procedures, and guidelines. None of the deviations raised in this review precluded entry into LTO.

9.1.2 Outcomes of the Global Assessment

The main aim of the global assessment (GA) was to determine the level of the overall safety of the plant by assessing, among other things, the combined effects of all safety factor review outcomes. It also provided the information required to produce the justification for continued operation until the next PSR and the suitability of the plant for continued operation for the intended period of LTO.

This was achieved by having an expert team cross-examine the impact of the negative findings (deviations) and positive findings (strengths) both in terms of an individual and a cumulative impact on plant safety. Safety assessments relating to fundamental safety functions (FSF), levels of defence in depth (DiD), and IAEA fundamental safety principles (FSP) were used to inform the conclusions relating to the continued operation.

The safety factor review outcomes were deemed to be comprehensive and provided sufficient data for use in the GA process. The GA process provided a balanced view of the cumulative effect of the deviations on DiD, DSA, FSFs, FSPs, probabilistic safety assessment (PSA), the emergency plan (EP), and ageing management (AM), and it could be concluded from the GA outcomes that the overall safety of the plant is adequate and that the plant is suitable for continued operations. The GA process produced the necessary safety improvements to maintain risk within acceptable levels, ensure continuous improvement, and support safe operation over the next PSR period and into LTO.

Based on this comprehensive PSR, including the outcomes of the safety factor reviews and the GA, the assessment of the suitability of the plant for continued operation [115] concluded the following:

- While the safety factor reviews identified 113 deviations and the global assessment identified eight global issues that required resolution within the time frame commensurate with the deviation safety significance, all the safety factor reviews concluded that it is safe to continue to operate. The overall nuclear safety performance of the facility was, thus, at an acceptable level.
- The safety factor reviews that specifically covered aspects of LTO concluded that there are no challenges to LTO, provided the deviations linked to LTO were resolved timeously. In particular, the following conclusions could be drawn from the reviews:
  - The current design of the plant is adequate when assessed against the licensing basis and national and international standards. The plant design processes and procedures are adequately robust to maintain the ongoing integrity of the plant design and safety case.
  - The programmes associated with maintaining the condition of the SSCs are adequate and well implemented. The actual condition of the SSCs important to safety provides confidence in the delivery of safety functions until the next PSR including LTO.
  - The equipment qualification programme is well aligned with international standards and capable of ensuring qualified equipment throughout LTO.
  - The ageing management programmes, processes, and management method requirements are largely met, and LTO could be achieved with the proposed enhancements.
  - The hazards (internal and external) are understood, and there are means to mitigate the hazards.
  - An integrated management system, in line with international standards, had been implemented and included a comprehensive quality assurance programme.

  * The human resources processes and procedures are well documented and in line with international standards. A workforce plan is in place that provides for sufficient staff for safe operation and LTO.

- The 14 strengths identified have no mitigating impact on the deviations; however, the strengths do provide evidence that examples of organisational excellence exist. An example is the strength in the area where international operational experience could support the facility in continuing safe operation, including LTO.
- The outcome of the GA supports continued safe operation, including LTO. In particular, the following conclusions were drawn from the analyses:
  * The FSP requirements are met and would likely continue to be met.
  * The cumulative effect of deviations on the available provisions for defence in depth is not of significant concern, and none of the deviations required regrading.
  * The overall risk of the current plant, without any new safety improvements, remains acceptable, as no new deviations were identified that required immediate justification for continued operations.
  * The FSFs are not significantly affected by the cumulative effect of the deviations and remain intact.
- Mitigations are in place for the CRE in-leakage, and safety improvement actions have been proposed, which would be implemented within timescales commensurate with the risk.
- The cumulative increase in PSA Level 1 risk for internal events was approximated at 6E-07/y, and the cumulative increase in Level 2, Level 3, and spent fuel pool (SFP) PSA was found to be no more than minimal.
- The overall impact of the external events deviations on the PSA could be conservatively categorised as a "medium" risk to the plant and the risk will be reduced once the safety improvements are implemented.
- The DSA review concluded that several deviations were identified as having an impact on DSA. Based on the arguments provided, adequate compensatory measures exist.
- It was determined that the deviations do not have a significant impact on the ability to execute the EP and that there is no need for an immediate review of the emergency plan technical basis (EPTB). The EPTB was, however, being reviewed for LTO.
- Although the plant is not yet fully aligned with the international requirements related to DEC, the deviations do not have a significant impact on the existing ability of the plant to prevent and/or mitigate DBA, DEC-A, or DEC-B accidents.

- The current licensing basis largely remains valid, and it was likely that the facility would continue to meet its licensing basis for the duration of LTO, with the implementation of the PSR IIP.
- All deviations identified have suitable safety improvement actions, and the timescales for their implementation are considered appropriate and commensurate with the deviation's safety impact. These were included in the PSR IIP, which was informed by the risk to the plant using PSA and deterministic methods. In addition, the analysis performed in the PSR global assessment confirmed that the overall risk of the current plant, before any new safety improvements are implemented, remains acceptable.

Therefore, the suitability for continued operation (SCO) assessment concluded that the outcome of the PSR supports continued safe operation and that LTO is feasible, with the implementation of the safety improvements in the PSR IIP [115].

9.1.3 PSR Safety Improvements

The PSR IIP is the culmination of the global assessment process and represents the safety improvements to be undertaken in support of the case for SCO. The IIP represents the safety improvements required to enhance or maintain nuclear safety levels through a set of specific interventions.

The PSR IIP was compiled utilising existing station processes (adapted to the PSR, where necessary). The IIP also considered the NNR commitments related to the PSR ranking of safety improvements for the prioritisation and categorisation of the safety improvement actions in the IIP, ensuring organisational alignment in terms of the proposed resolutions and the timescales for implementing the work activities.

Safety improvement actions and assigned due dates were reviewed and agreed on by the manager responsible for implementing the safety improvements and endorsed by the Nuclear Safety Review Committee (NSRC) and the Nuclear Executive Committee (NEXCO). The Eskom Chief Nuclear Officer endorsed the PSR IIP for submission to the NNR. Progress on implementation of the IIP actions will be monitored through the corrective action programme (CAP) and reported to the NOU executives on a six-monthly basis.

The development of the PSR IIP and the strategy for implementation is described in Appendix I of 331-608, [115].

9.2 Compliance with Safety Criteria and Requirements

Koeberg has implemented programmes to ensure that operations adhere to safety criteria and requirements. The criteria and requirements are included in the Koeberg safety performance indicators, and compliance is monitored continually to ensure safe operation. Compliance is regularly reported to the NNR through the quality assurance monitoring programme and licence-binding surveillances, and in the event of non-compliance, these are captured in the CAP database for history recording and resolution tracking.

9.2.1 Principal Safety Criteria

The regulations on safety standards and regulatory practices (R.388) [241] specify the regulatory requirements applicable to holders of nuclear authorisations. The requirements on risk assessment and compliance (RD-0024) stipulate the principal safety criteria, which refer to limits on the annual risk/dose to members of the public and workers because of exposure to radioactive material resulting from accident conditions or normal operations [289]. The 'Koeberg Nuclear Installation Licence' (NIL-01 Variation 19) [286] stipulates the conditions with which the plant must comply for the safe operation of the facility. RD-0022 (*Radiation Dose Limitation at Koeberg Nuclear Power Station*) [288] specifies the radiation dose limitation for both normal operating conditions and during emergencies at Koeberg.

The PSR review of the probabilistic safety assessment (PSA) confirmed that the safety criteria and requirements had been respected and since the LTO does not result in a significant radiological risk increase, it will continue to be respected in the LTO period [65]. Although, in recent years, the risk profile had shown an increase due to recently adopted operations such as casking the assessment concluded that the overall PSA results demonstrated that the risk associated with the operations of the plant are well within the principal safety criteria (that is, risk limits) specified in RD-0024 (*Requirements on Risk Assessment and Compliance with Principal Safety Criteria for Nuclear Installations*) [289] and, therefore, supported safe continued operation. The PSR also reviewed the deterministic safety criterion through the review of the radiation protection programme, and the outcome of the review confirmed that the plant meets this criterion, which supports continued operations. Further details to justify the assertion mentioned above are discussed below.

9.2.2 Radiological Risks

The facility complies with the risk limits in the regulation for safety standards and regulatory practices (R.388) [241]. The PSA risk profile is quantified in the risk assessment report (RAR), PSA-R-T19-01 [225]. The risk profile is presented in terms of the peak public risk, average public risk, peak site personnel risk, and average site personnel risk. The core damage frequency (CDF) and the large early release frequency (LERF) for Koeberg are well below the IAEA-recommended limit for the existing plant. (Refer to Figure 9-1 and Figure 9-2.)

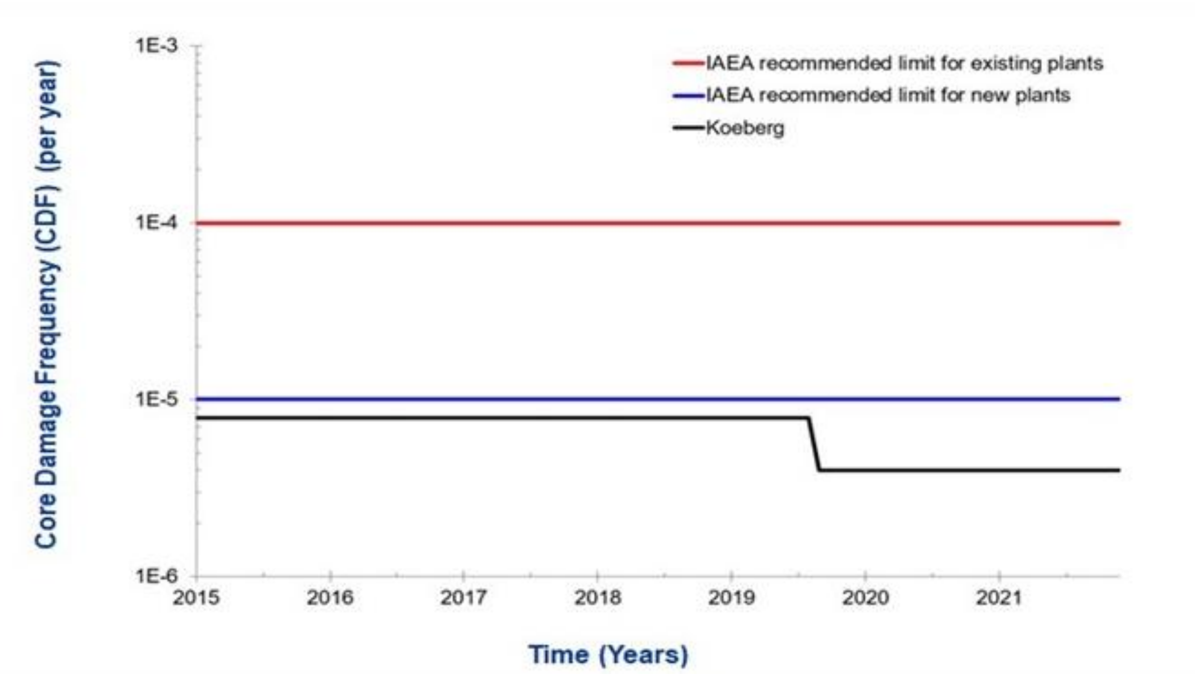

Figure 9-1: *CDF per year at KNPS from 2015 to June 2021*

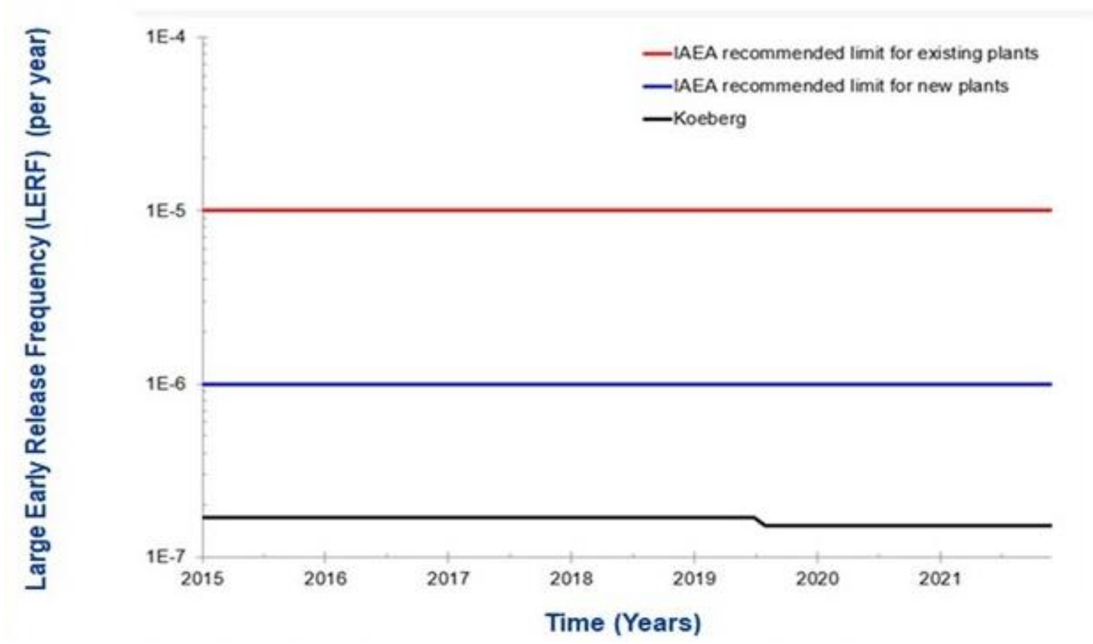

Figure 9-2: *Large Early Release Frequency (LERF) from 2015 to June 2021*

In 2019, an increase in the peak public risk was observed due to the risk associated with the drop of a cask loaded with fuel during casking operations. However, the increase in risk remained within the NNR limit. Refer to Figure 9-3.

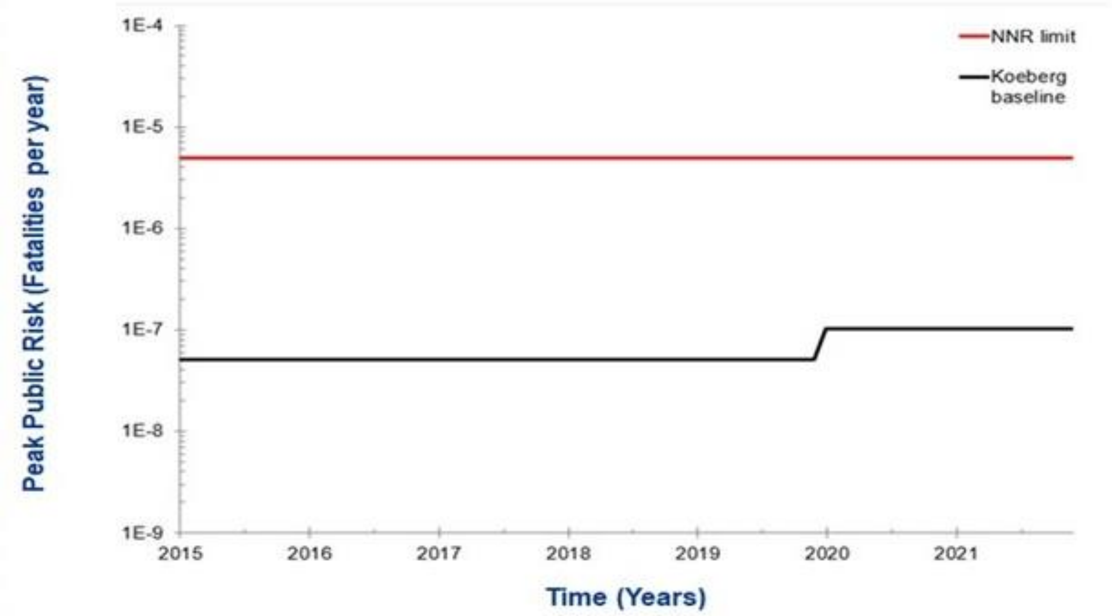

Figure 9-3: *Peak Public Risk, including Risk Associated with Cask Loading; 30 June 2021*

The risk profile is submitted to the NNR every six months. Bounding assumptions in the PSA regarding the screening of accidents relevant to worker risk are not affected by LTO. These assumptions relate to radiological source terms originating from fuel handling accidents, reactor accidents, spent fuel pool accidents, and cask accidents used in calculating worker risk.

With regard to the installation of the new steam generators, the source term for irradiated fuel in the reactor has been updated for operations and will remain applicable and bounding for the proposed LTO period.

To assess the impact of spent fuel pool accidents on on-site personnel, the PSA conservatively assumes that at all times the spent fuel pool is fully loaded with spent fuel assemblies. This assumption remains valid and enveloping for the LTO period and future decommissioning.

The PSA *(Risk Assessment of Additional Metal Casks)* [222] conservatively assumes a probability of cask seal failures and aircraft crash accidents that can affect all 161 casks. This number of casks envelopes the additional 20 years of LTO and complete unloading of the spent fuel pools during decommissioning.

The Koeberg SAR III-4.4 (*Radiological Consequences of Accidents*) [178] provides an overview of the radiological consequences of design-based accidents. Public dose analyses were performed for the SG replacement project [1] using updated assumptions (with appropriate conservatism), modelling, and methodologies such as the introduction of the reference core and alternative source term methodology, in accordance with 331-195 (*Koeberg Accident Analysis Manual*) [102]. The updated analyses demonstrated compliance with the public dose acceptance criteria for design basis accidents with significant margins.

Radiological consequences to the public from nuclear accidents, off-site transportation of radioactive material, and emergency plan provisions for the protection of the public during emergencies were reviewed in the PSR‘s DSA, PSA, safety performance, and emergency planning safety factors. The PSR concluded that Koeberg complies with selected national and international requirements, standards, and guidance related to radiological consequences to the public from nuclear accidents. Few deviations related to this subject have been identified and associated safety improvements have been prioritised for resolution in accordance with the schedule contained in the PSR IIP [115]. These safety improvements relate to:

- the development of external hazard PSA;
- consideration of steam line break (SLB) – induced steam generator tube rupture (SGTR) in the PSA model;
- inclusion of internal event grouping in the PSA for casks and the possibility of damaging SEC pipework during cask transfer to and from the fuel building;
- inclusion of fire and flooding initiators in the PSA model;
- inclusion of all environmental and phenomenological conditions in the PSA model;
- update of the PSA model with the most recent plant-specific reliability data; and
- update of PSA Level 3 with the most recent population data.

The peak and average public risks are quantified in the RAR [225]. The assumptions in the PSA regarding the peak public risk are not affected by changes to population data during LTO, as the peak public risk refers to the individual risk that a member of the public may accrue at the site boundary. Any changes to the operations or site boundary that may affect the peak or average public risk will be evaluated and submitted to the NNR for approval in accordance with the PSA update process.

The assumptions in the PSA regarding average public risk may, however, be affected by changes to population data during LTO. Changes in the local and national population numbers affect the calculation of the average public risk in accordance with the methodology in the RAR [224]. Extrapolation of population data from the 2011 census indicates that a sufficient margin for the average public risk remains below the relevant NNR safety criteria and requirements until the end of the LTO period [220], [225]. Eskom has performed an evaluation of outdated population data used in the PSA studies and obtained concurrence from the NNR to the use of scaling of average public risk in the interim. The use of the scaling factors for the average peak public risk is documented in the impact evaluation of outdated population data used in the PSA studies report, PSA-R-T-16-22 (*Impact Evaluation of Outdated Population Data Used in PSA Studies*) [223].

The PSR PSA review [115] of the radiological risk to site personnel and the public concluded that the plant complies with national regulations, and international standards and guidance. According to regulatory safety criteria and requirements, the peak and average site personnel risks quantified in the RAR [225] demonstrate that workers are protected against undue deterministic and stochastic radiation health risks.

The requirements to regularly update the PSA assessment of public risk during the LTO and decommissioning period to quantify any public risk changes exist. These requirements are documented in 331-33 (*PSA Updating and Maintenance*) [109]. Protection of the public against undue stochastic radiation health risks during the LTO period will, therefore, be ensured through continued compliance with the relevant regulatory safety criteria and requirements.

9.2.3 Current Licensing Basis

The current licensing basis provides the nuclear regulatory requirements and the licensing documentation that must be adhered to during operations to ensure that the safety criteria and licence conditions are always respected. The PSR comprehensively reviewed the validity of the current licensing basis, the adherence of the plant to the applicable regulations and Regulator guidelines, and the adequacy of licence-binding documents. The PSR concluded that the current licensing basis of Koeberg remains valid, and once the DSSR studies are concluded in 2024, their impact on the current licensing basis will be reviewed.

The PSR verified the validity of the current licensing basis, and a medium-graded deviation raised was that the specific site characterisation has not been concluded, and thus the site safety report has not been updated with the latest information relating to external events applicable to the facility. The studies for the update of the specific site characterisation are currently underway, and scheduled to be completed prior to entry into LTO. Further details on these studies and the justification for safe LTO are provided in § 9.3 of this report.

The PSR identified inadequacies related to partial compliance with some LTO regulatory guidelines [294] associated with the requirements for safety-related programmes. Low graded deviations were raised for these inadequacies and the necessary safety improvements were prioritised accordingly to meet the requirements for LTO (that is, safety improvements categorised into those required prior to entry into LTO and those that would be performed during the LTO period).

All the deviations were considered in the PSR global assessment to determine the impact of these deviations on the continued safe operation of the plant and to determine the relevant safety improvements. These safety improvements are scheduled accordingly either in the LTO integration preparation plan or in the LTO implementation plan in § 14.0.

9.3 Specific Site Characterisation

LTO requires a demonstration that the site-specific characteristics have been comprehensively reviewed for applicable hazards important for the safe operation of the installation and the site safety report updated to reflect changes in the hazards. This section discusses the external hazards applicable to the site. A change in the applicable hazards may potentially affect some of the facility's current licensing basis documents (such as the SAR, design basis, etc.). Considering the potential changes in the current licensing basis, this section excludes the evaluation of the robustness of the design to withstand changes in the hazards and the reassessment of the design basis. These exclusions are discussed in the plant design in section § 9.4.

The Duynefontyn site characteristics have been assessed over the years to determine hazards that can affect the safety of the facility. The recent studies of the site-specific characteristics have been completed, except for the seismic hazard analysis and a detailed probabilistic tsunami hazard assessment. These remaining two studies are ongoing and will be completed in 2024 before the end of the current licence term.

The impact of the site-specific hazards on the design of the plant is discussed in section 9.3.1.3. In the case of new hazards (that is, hazards not considered in the initial characterisation) or changes in current hazard parameters, the potential risk has been assessed for impact on safe LTO. The site continues to be suitable for the facility and has been characterised considering changes in hazards, and these changes do not impede safe LTO.

The site-specific external hazards and site conditions have been re-evaluated under the DSSR project, considering advances in knowledge, and understanding of external events and changes to regulatory requirements. The scope of the DSSR is informed by the requirements of Regulation No. R.927 (*Regulations on Licensing of Sites for New Nuclear Installations*) [243]. The content is in accordance with sections 4 and 5 of R.927, as well as RG-0011 (*Interim Guidance for the Siting of Nuclear Facilities*) [292]. Although the regulations on siting apply to new nuclear build, it was considered that the aspects of site characteristics are relevant to Koeberg. The DSSR Chapter 5 (Site Characteristics) [216] defines the technical basis for Chapter 2 (Site Characteristics) of the SAR [178] and also serves as input into any potential reassessment of the plant design basis.

Below is a summary of the site-specific characterisation studies' outcomes, which support the assertion mentioned above related to safe LTO. The PSR hazard analysis reviewed all natural and human-induced external hazards and site conditions. This review considered operating experience and new safety-related information, and the results are discussed in § 9.4.6.2.

9.3.1 Summary of the Site Characterisation Studies

All external hazards (natural and human-induced) have been identified and evaluated, except for the seismic hazard evaluation and a detailed probabilistic tsunami assessment, which are in progress. The extent of the evaluation was commensurate with the safety significance of the potential hazard at the site based on the initial screening. (Refer to Chapter 6 of the DSSR.) The hazards were evaluated into the following three screening categories:

- Screened out unconditionally on the basis of being incapable of posing a physical threat or being extremely unlikely with a high degree of confidence

- Screened out conditionally (a measure of uncertainty exists, and design confirmation or further studies are required)
- Screened in (impact on the design to be assessed)

9.3.1.1 Hazards Screened Out Unconditionally

The following hazards are screened out unconditionally (additional information is contained in Chapter 6 of the DSSR [216]):

- Collapse, subsurface movement, or uplift of the site surface

  The event is of equal or lesser damage potential than the events for which the facility was designed. The event cannot occur close enough to the facility.

- Slope instability

  The event is of equal or lesser damage potential than the events for which the facility was designed.

- Meteorological events, excluding extreme winds (hurricane-force winds and tornadoes)

  The event has a significantly low mean frequency of occurrence when considering regulatory target safety goals, taking into account the uncertainties in the estimates, where available data permit.

- External flooding from terrestrial sources (water-retaining structures and rivers)

  The ground may become temporarily waterlogged in limited areas after intense periods of precipitation, but there are no hydrological features that would present a safety problem from stream flow or flooding from adjacent properties. The average vulnerability and safety consequences are low, which include the still high-water boundary condition from the sea. The still high-water levels (4,49 m above mean sea level (MSL), 5,30 m above MSL, and 6,19 m above MSL for a 1E-4, 1E-6, and 1E-8 annual probability of exceedance, respectively) are below the existing main terrace of approximately 8,0 m above MSL.

- External fires

  Mitigation against the occurrence of veld fires resulting in air pollution is included in nuclear installation design. In addition, existing management of the site vegetation that is exercised in respect of the Koeberg and transmission servitudes is mitigation against veld fires.

- Hazardous materials – land-based stationary and transport sources (off-site sources)

  Hazardous materials are screened out due to screening distance.

- Electromagnetic interference

Electromagnetic interference is dealt with in the design and operations of the facility as well as the security protocols at Koeberg.

9.3.1.2 Hazards Screened Out Conditionally

The following hazards are screened out conditionally:

- Soil liquefaction

  A wide distribution of soils investigated on site has had a high liquefaction potential, with a notable exclusion of the facility nuclear island, under which liquefaction potential was eliminated using cement-stabilised soils during construction of the plant.

- Biological phenomena and related events

  The potential impact of marine organisms on the cooling water supply can be dealt with through appropriate design and management measures.

- Loss of freshwater supply

  The facility is supplied with potable water from the municipality; however, given the impact of drought as was experienced from 2015 to 2018, reliance on municipal water supply cannot be guaranteed. Based on the scarcity of conventional local and regional water supplies in the site region, it is proposed to augment potable water with a backup system of groundwater from the Aquarius wellfield supplying a desalination plant at Koeberg. Desalination of seawater offers the best short- to long-term option for the site. The loss of freshwater supply to Koeberg in the short-term is low since the priority of the municipality is to supply Koeberg due to its national key point status.

9.3.1.3 Hazards Screened In

The section below describes all the hazards associated with the external events that are screened in as applicable to the Duynefontyn site. The description includes an indication of any changes to current hazard parameters and/or identification of new hazards. These are the hazards that are utilised to evaluate the robustness of the design of the plant. The details of the screening reports are contained in the DSSR [216].

1. Earthquake-induced ground shaking and surface faulting

   The initial geological, seismological, and geotechnical site safety studies performed found no indications of surface rupture. Eskom is conducting confirmatory studies, which include a probabilistic seismic hazard analysis (PSHA), in accordance with the enhanced Senior Seismic Hazard Analysis Committee (SSHAC) Level 2 guidelines, followed by a seismic probabilistic safety assessment. These studies are in progress and are due for completion in 2024, prior to entry into LTO. These studies are included in the LTO IPP in Appendix A.1.

In support of safe LTO demonstration, a baseline seismic hazard analysis [127] was conducted and a baseline seismic curve for the site was developed. While the baseline seismic hazard analysis was based on the most comprehensive and objective assessment of available data, models, and methods to date, the large epistemic uncertainties in the seismic source model and the ground motion model mean that the results are likely conservative. These results will be refined during the performance of the SSHAC-enhanced Level 2 study.

The outcome of these baseline studies indicates an increase in peak ground acceleration compared to the original design basis safe shutdown earthquake data (that is, the site was designed for a peak ground acceleration of 0,3 g with some margin, while the preliminary studies indicated that the new peak ground acceleration was greater than the design basis peak ground acceleration), thus requiring further evaluation of the robustness of the plant against this hazard to justify LTO as discussed in § 9.4.6.2. § 9.4.6.2 provides further details on the plant robustness assessment performed because of the changes in this hazard.

2. Water quality

   Corrosion risk to foundations is considered low. The facility has implemented a groundwater monitoring programme to monitor the corrosion risk to foundations even during the LTO period.

3. Flooding from the sea, including tsunamis

   Flooding from the sea due to the following was assessed:

   - Tsunami run-up combined with sea level rise, high tides, and positive storm surge
   - Storm wave run-up combined with sea level rise, high tides, positive storm surge, wave set-up, and basin seiche

Tsunami hazard

The tsunami hazard included in the KSSR was based on a magnitude 7,8 seismic upheaval at the South Sandwich Islands. In the KSSR, an estimated tsunami run-up of +4,0 m MSL was envisaged, with a maximum credible tsunami of +5,2 m MSL if combined with the highest astronomical tidal level.

The updated tsunami hazard assessment (THA) considered all probable sources, that is, near-field and far-field sources. The results showed that the probable maximum tsunami (PMT) run-up and inundation were governed by volcanic flank collapse tsunamis. No other tsunamigenic sources, including distant earthquakes and local submarine landslide sources were identified to be of concern. The source of volcanic flank collapse risk is Tristan da Cunha, a volcanic island approximately 3 000 km west of the Cape, which has evidence of a flank collapse between 6 000 and 35 000 years ago. In the unlikely event of a tsunami, the current conservative estimates for PMT, considering climate change and other factors, range from a maximum of +11,82 m to +13,95 m between 2021 and 2064.

A preliminary tsunami probabilistic analysis (TPA) for a volcanic flank collapse on Tristan da Cunha (the dominant risk) was conducted to understand the risk to the plant. The study concluded that the probability of occurrence is low and therefore the risk to plant is low, and provisions exist in the design extension domain to mitigate the impact in the unlikely event of a tsunami. However, a detailed TPA to confirm the preliminary findings will be performed. This analysis is included in the LTO IPP and scheduled for completion prior to LTO.

A focused palaeotsunami field investigation of sites that were judged to be favourable for recording tsunami deposits near Duynefontyn was carried out with the aim of identifying and assessing the magnitude of any recent prehistoric tsunamis. No evidence supporting palaeotsunamis in the area was identified. However, the onshore geological record is of relatively short duration and incomplete, even for the late Pleistocene (~3 000 to 11 000 years), given that, for much of that time, the sea level was below the present level, and tsunami deposits would likely be eroded during multiple sea-level fluctuations, including the most recent post-glacial transgression from the Late Glacial Maximum (20 000 to 21 000 years).

The PSHA results will be reviewed to verify its impact on the THA. The verification will be performed on completion of the seismic hazard analysis prior to entry into LTO. Should the assessment indicate a high risk, appropriate mitigations will be put in place.

Storm wave run-up

Recent studies indicated that the likelihood of a storm wave run-up breaching the terrace has increased. While the original terrace design height was based on a 1E-06/y return frequency, the latest data indicated that the return frequency of exceeding the terrace height is now between 1E-04/y and 1E-06/y. However, the impact was concentrated on the north and south of the nuclear terrace and did not affect SSCs. Only at 1E-08/y did the wave run-up flood the terrace adjacent to the reactor buildings. Therefore, the likelihood of a storm wave run-up exceeding the terrace level and having an impact on the facility is considered low.

4. Coastline erosion

The coastline stability was evaluated by measuring the horizontal distance from the baseline to the most landward extent where any erosion or accretion was observed on the profiles. The model was run for extreme storms with exceedance probabilities of 1E-02/y, 1E-04/y, 1E-06/y and 1E-08/y. The model was run for the following dates to include the effect of climate change on waves, water levels, and coastline stability:

- 2021: present-day.
- 2064: end of decommissioning period.

The predicted erosion lines for different probabilities are presented in section 5.9 (Oceanography) of the updated DSSR [216], and the results are summarised in Table 9-1. The results show that coastline erosion increased over time due to long-term coastline trends, sea level rise, and larger waves. In 2021, the erosion line south of Koeberg was predicted to reach the root of the revetment protecting the outfall structure at 1E-02/y. The predicted long-term accretion south of Koeberg reduced the risk to the outfall structure by 2064. In 2021, it was predicted that the erosion line north of Koeberg would reach the root of the northern breakwater at 1E-08/y, with this probability increasing to 1E-02/y by 2064.

For the 1E-02/y and 1E-04/y events, the rate of erosion because of sea level rise would not pose an immediate risk of damage to the breakwater and intake structures. The maximum coastline erosion occurred on the northern side of the site, except for the 1E-08/y storm, where the dune ridge was breached south of Koeberg, but the probability was low.

Table 9-1: *Maximum Coastline Erosion Adjacent to Koeberg.*

Exceedance Probability (per Year)	Total Coastline Erosion Adjacent to Koeberg (m from Baseline)	
	2021	**2064**
1E-02	-59	-145
1E-04	-74	-159
1E-06	-87	-175
1E-08	-286	-306

Regular surveys have been performed to monitor long-term erosion, and accretion of the beach and seabed in the vicinity of the cooling water intake basin, and breakwater structures. The surveys will continue during the period of LTO.

5. Extreme winds, including tornadoes

Tropical cyclones

Recent studies indicated that the site region is not on a hurricane (tropical cyclone) track or adjacent to a warm ocean. Therefore, it is not expected that the site would experience a cyclone. Tropical cyclones are generated in areas where the ocean surface temperature is greater than 27 °C and between latitudes 5° S and 30° S. The site is located south of 33°S and is, therefore, not subject to tropical cyclones.

Although the site is not prone to hurricanes, hurricane-force winds are said to have occurred at the site. According to the Beaufort scale, a wind speed scale, the term “hurricane-force winds” refers to winds with speeds above 118 km/h (32,8 m/s). This wind speed (as a gust) has been exceeded five times (1986, 1987, 1993, 1994, and 2002) over the 40-year monitoring period at the site. The highest gust of 38,8 m/s occurred in 1987. The measured highest hourly average wind speed for the site for the 40-year monitoring period (1980 to 2019) was 17,2 m/s.

From the analysis of wind gusts observed at the 10 m level of the 120 m tower for the period 1980 to 2020, the wind gust corresponding to a return period of 100 years was 43,6 m/s, 1 000 years was 53,0 m/s, 10 000 years was 62,4 m/s, 100 000 years was 71,9 m/s, and 1 000 000 years was 81,3 m/s. The maximum mean hourly velocities for these return periods were 21,9 m/s, 26,7 m/s, 31,5 m/s, 36,3 m/s, and 41,1 m/s, respectively. The Koeberg design basis wind for Class 1 buildings is 62,5 m/s (maximum 3 s gust), and the design basis wind speed for buildings other than Class 1 is 38,3 m/s (maximum mean hourly velocity). The actual measured extreme wind speeds at Koeberg over the 40-year period have not exceeded the design basis wind speeds for the Class 1 or non-Class 1 Koeberg buildings.

The IAEA safety standard SSG-18 (*Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations*) [263] recommends using the 3 s gust wind speed at 10 m above the ground that has a 1% annual frequency of exceedance (100 year mean recurrence interval) to specify wind loads. The design basis wind speed (3 s gust) for Class 1 buildings at 62,5 m/s is only exceeded at a return period of 10 000 years and for non-Class 1 buildings (maximum mean hourly velocity) at a return period of between 100 000 and 1 000 000 years. Extreme winds (cyclones), therefore, present a low risk to safe operations at Koeberg.

Tornadoes

Tornadoes and hurricane wind speeds were not considered in the KSSR Rev. 0. A recently conducted meteorological study indicated that the tornado activity had increased since 1987 within an 80 km radius from the site. The estimated average tornado strike frequency for the site region (regardless of the severity) was calculated to be 1E-05 per year per km^2. The frequencies per severity are given in Table 9-2 below.

Table 9-2: *Tornado Frequencies per Severity*

Meteorological Parameter			Value	
Tornadoes	Tornado probability (EF-enhanced Fujita scale)	Based on a 116-year database 1905 to 2020	All	1E-05/y per km²
			EF0	7E-06/y per km²
			EF1	2,4E-06/y per km²
			EF2	5,6E-07/y per km²
			EF3	1E-08/y per km²
			EF4	<1E-08/y per km²
		Based on a 34-year database 1987 to 2020	All	2,2E-05/y per km²
			EF0	1,7E-05/y per km²
			EF1	5,2E-06/y per km²

Meteorological Parameter			**Value**	
			EF2	1,2E-06/y per km^2
			EF3	2,2E-08/y per km^2
			EF4	<2,2E-08/y per km^2
	1E-07/y wind speed: - maximum translational - maximum rotational		 75,0 m/s 60,0 m/s	
	Path width: - F1 tornado (80% probability) - F2 tornado (19% probability) - F3 tornado (1% probability)		 10 m to 50 m 30 m to 400 m 50 m to 1 100 m	
	Pressure drop for 1E-07/y wind speed		40 hPa	
	Maximum rate of pressure drop for 1E-07/y wind speed		13 hPa/s	

From the table above, EF2 tornadoes have an expected probability above 1E-07/y and EF3 tornadoes well below 1E-07/y. Using the EF scale, the estimated maximum tornado wind speeds (three-second gust estimates) are between 61 m/s and 75 m/s. This exceeds the facility design basis three-second gust velocity of 62 m/s.

Design considerations regarding winds are wind loading on walls and missile protection. Weaknesses in design related to this hazard were identified during the external events safety reassessment (EE-SRA), which concluded that further studies related to the hazard were required and also identified design vulnerabilities (such as the emergency diesel generator (EDG) radiators and some ventilation intakes) that needed enhancements. Regarding the identified design vulnerabilities, modification 12029 (*Tornado and High Wind Enhancements*) was, therefore, raised. Refer to letter K-28627-E submitted to the NNR.

6. Aircraft crash

Types of aircraft are classified into civil aviation and military aviation. Civil aircraft contains two major categories, namely, commercial aviation and general aviation. These categories are subject to different flight path restrictions and regulations. The annual aircraft crash rate was determined for each aviation category by considering airfield-related events from nearby airports (landing and take-off activities), in-flight events (directly over or in the immediate proximity of the site), and background crash rates (random crash rates within South Africa).

The aircraft crash rates were calculated per unit site area and are independent of the dimensions of existing or planned plant buildings located at the site (see Table 9-3 below). Table 9-3 provides the results of the assessment of the potential aircraft crash risk as contained in the DSSR.

Table 9-3: *Current Expected Annual Aircraft Crash Rate for the Duynefontyn Site*

	Aviation Category			Estimated Crash Frequency (per y/km²)
Current expected annual aircraft crash rate per km² for the site	Civil aviation	Commercial aviation		7,46E-07
		General aviation	Fixed-wing	3,24E-04
			Helicopters	9,34E-07
	Military aviation	Large		1,53E-07
		Small		2,37E-07
		Helicopters		2,92E-05

Results from the aircraft crash study showed that fixed-wing civil aviation and military helicopters have occurrence frequencies greater than 1E-06 per year per km^2. The risk to the site is considered low, as the immediate airspace above the site is a registered restricted flying area (FAR36 – Restricted Area). The area covers a footprint with a 4,63 km radius, with its centre at 33°41’00S and 18°26’50E, excluding areas east of the R27 road. The restricted area extends from ground level to 610 m above mean sea level (aMSL). To the east of the site is a danger area (FAD 200A – Danger Area) from ground level to 610 m aMSL, and north-east of the site is a danger area (FAD 200B – Danger Area) from ground level to 1 220 m aMSL.

7. Extra-terrestrial events

The primary concern regarding extra-terrestrial events such as solar storms is the risk of an extended loss of off-site power. The risk has been mitigated by the mobile emergency backup diesel generators.

It is demonstrated above that, although there have been changes in some hazard parameters and new hazards have been screened in, the changes in conditions (with the exception of the seismic and TPA, for which interim studies were conducted, while the detailed studies are still in progress) do not pose a risk to LTO. The impact of the seismic hazard on the screened-in hazards will be confirmed when the seismic studies are completed prior to entry into LTO.

9.4 Plant Design

The section describes aspects related to plant design and demonstrates the adequacy of the plant design for LTO. Related to design aspects, consideration was given to the following:

(i) The current plant design, including applicable changes since the commissioning of the plant

(ii) The design basis and design criteria

(iii) Application of defence in depth

(iv) Fulfilment of the fundamental safety functions

(v) Design safety margins and safety analyses

(vi) Configuration management and design documents

In addition to the aspects mentioned above, the following elements, mainly related to operating experience, were considered to further justify the adequacy of the design:

- Challenges to LTO from the USA life extension operating experience
- Operating experience guidance from EPRI
- Gaps found in the licence requirements
- Maintaining the robustness of the three protection barriers
- Provision for ageing management in the design

The current design of the plant is deemed adequate for long-term operation when assessed against the licensing basis and national and international standards based on the conclusions of the PSR plant design review and the global assessment report, except for the design of the control rooms. The design of the control rooms did not meet the design basis in SAR II-4.5.1 (a design basis that is based on the general safety criteria of US NRC 10 CFR 50 Appendix A General Design Criteria No. 19) and, therefore, was not deemed adequate to protect the operators against radiation exposure during accident conditions. This design deficiency is further discussed in section § 9.4.2.1.3.

Plant design processes and procedures are adequate to maintain the integrity of the plant design and its documentation to support safe LTO. The PSR assessment of the design documents and configuration management, which found no deviations related to these design aspects, demonstrated this assertion. The plant design processes and procedures provide confidence that modifications to the plant will not affect plant safety for LTO. It was demonstrated through the PSR that Koeberg has appropriate processes to manage any future design issues that might arise during the LTO period. The PSR plant design review concluded that no deviations related to plant design precluded continued plant operation or safe LTO.

The description of the plant design and the design basis are documented in the SAR [178]. Therefore, the description of the design in this section will be limited to the aspects related to pressurised water reactor (PWR) technology that directly address radiological barriers.

9.4.1 Results of the Plant Life Extension Feasibility Study

The feasibility study concluded that the extension of the life of the plant by an additional 20 years was feasible, provided that the identified life-limiting major components were replaced. Implementing these replacements is included in the scope of preparation activities for LTO.

The feasibility study found that the following components would require replacement:

- Steam generators due to primary water stress corrosion cracking, and the replacement is due in 2024 (the activity is included in Appendix A.1).
- Refuelling water storage tanks due to atmospheric stress corrosion, and the replacement has been completed.
- Reactor vessel head due to primary water stress corrosion cracking, and the replacement has been completed.
- Partial replacement of the high- and low-pressure feedwater heaters due to erosion, fretting, and water hammer. Although these components were mentioned in the feasibility study, they are not discussed further in the safety case, as they are not SSCs important to safety. According to the life-of-plant plan, these components are replaced using a normal engineering life-cycle management process.
- Cables and switchgear replacements due to potential ageing effects. Subsequently, it was determined that large-scale replacement of these components is not required because industry operating experience later revealed that the ageing management philosophies and strategies are adequate to manage the degradation mechanisms and ageing effects. The details regarding the ageing management of these components are discussed in § 9.5.2.3.

9.4.2 PWR Design Features

The facility consists of two three-loop pressurised water reactors, each rated at 2775 MWth and designed for an assumed nominal operating lifespan of 40 years. The section discusses the adequacy of the plant design for LTO and is limited to SSCs related to radiation protection barriers. The PWR design utilises three physical barriers to prevent or minimise radioactive releases, namely:

(i) the fuel cladding;

(ii) the reactor coolant pressure boundary; and

(iii) the containment building.

These barriers ensure staff and public radiation protection under all design-based operating conditions. To help protect the structural integrity of the barriers, the design includes safeguard and safety systems consisting of two redundant trains with emergency backup power sources to prevent or mitigate the consequences of any design basis accident and design extension conditions. The design of these systems is detailed in SAR II-4.1 (*General Information Concerning Safeguard Systems*) [178]. In addition, the design of the spent fuel storage facilities is discussed since these facilities are another source of radioactive releases.

Koeberg was designed and built according to the standards and technology of the era, and the design of the plant has been continually improved through modifications identified through regular safety reassessments (PSRs) and normal life of plant plans, improved regulatory requirements and international standards, operational experience and lessons learnt, and for alignment with the latest technological advancements and insights.

Based on the elements considered above, some of the major components have been upgraded or will be upgraded, to ensure adequacy for LTO as discussed below. The LTO assessments have identified plant design changes or modifications to address design life limitations and equipment qualification limitations and to improve nuclear safety in preparation for LTO, and these are listed in Appendix A.

9.4.2.1 Safety-Related Design Description

9.4.2.1.1 The Radiation Barriers

- First barrier: fuel

  Zircaloy-4 alloy M5™ and the Westinghouse advanced cladding alloy ZIRLO™ were selected as the cladding material for the fuel (the first safety barrier) because of their favourable mechanical properties, such as high resistance to corrosion and low neutron absorption. Other materials used to construct the fuel assembly are Inconel 718 and stainless steel AISI 304L. The stresses on the cladding are limited by the choice of uranium oxide density, the pellet-cladding gap, and the initial helium pressure contained in the rods. The fuel assembly design incorporates intermediate mixer vane grids, which help minimise mechanical interaction effects (erosion by fretting) between the cladding and the grids to promote thermal-hydraulic exchanges between the channels.

  Considerable operating experience has been acquired with the Framatome M5™ and the Westinghouse Zircaloy and advanced fuel cladding ZIRLO™. Between 1989 and 2019, more than 5,4 million M5™ cladding fuel rods were operating in several reactors. The Westinghouse advanced alloy ZIRLO™ has likewise acquired a considerable amount of operating experience. The overall operating results of this type of fuel have indicated that the rate of cladding failure is considerably lower than the rate assumed for the design of the reactor coolant clean-up system. Further details regarding the operating experience and testing of the fuel are documented in SAR II-2.3 (*Mechanical Design of the Core*) [178]. Based on the operating experience documented in SAR II-2.3, this design and material choice are robust and used in many other plants and are, therefore, adequate for LTO.

  Figure 9-4 shows the number of fuel-leaking assemblies for each fuel cycle since Outages 111 and 211 for Units 1 and 2, respectively. Since the 21st-cycle outages, there has been a decrease in the number of fuel failures. This is indicative of the robustness of the design and material choice.

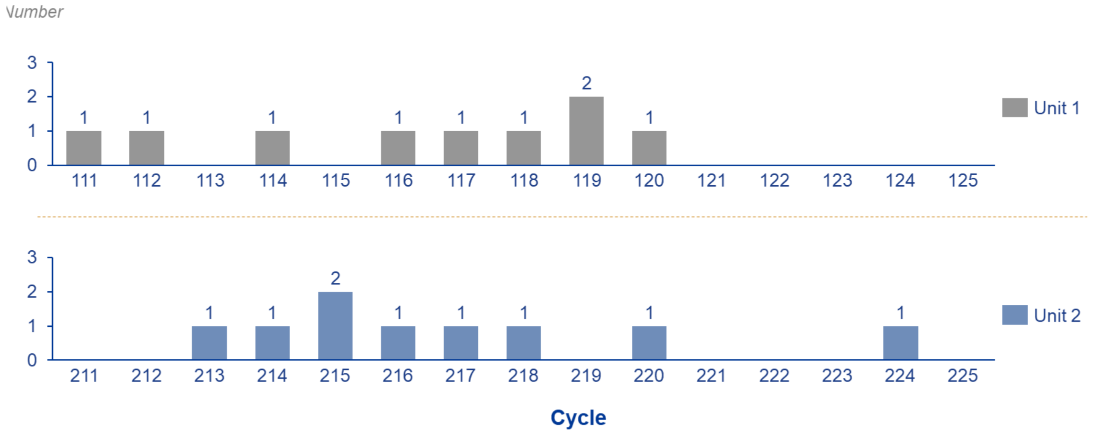

Figure 9-4: *Number of Fuel-leaking Assemblies for Each Cycle of Each Unit*

- Second barrier: reactor coolant pressure (RCP) system

  The reactor coolant pressure boundary is the second safety barrier to prevent the release of radioactivity. The RCP system and associated auxiliary control and protection systems are designed to ensure the integrity of the reactor coolant pressure boundary with adequate margins during normal operation and anticipated operational occurrences. The system is also designed to ensure core cooling and heat transfer from the fuel to the secondary system, contribute to reactivity control, and regulate the pressure of the reactor coolant. The RCP system is similar in design to the EDF 900 MW fleet. The detail regarding the design bases, safety function, and safety role of each major component and the interfacing of various other systems is described in SAR II-3.1 (*Reactor Coolant System Description*) [178]. The following major components from the second barrier were identified for replacement to ensure that the second barrier would be adequately robust for LTO.

  - **Steam generators:** the steam generators will be replaced (refer to Appendix A.1 LTO Integrated Preparation Plan) due to the material being susceptible to primary water stress corrosion cracking (PWSCC) (refer to modification 07092A (*SGR KNPS – Design Report – SGR System Design*) [1]. The new steam generators include improved tube material (Ni-alloy 690) that provides significant resistance to PWSCC and extended operational life. A comprehensive review and update of the associated design requirements and accident studies have been performed. Additionally, an updated fatigue transient listing has been established to cater for 60-year plant life.

  - **Reactor pressure vessel head:** the reactor pressure vessel (RPV) closure heads included Ni-alloy 600 penetration nozzles prone to PWSCC. This degradation mechanism led to the replacement of the Unit 1 reactor pressure vessel closure head in 2008 and Unit 2 in 2022. The new closure heads have Ni-alloy 690 penetration nozzles. This is the same design implemented in the EDF 900 Mwe fleet and numerous other international units. The construction material for the RPV is the American Society of Mechanical Engineers (ASME)

SA-508 Class 3, and the internals are layered with 7,5 mm-thick stainless steel (type 308L). The condition of the RPV material is monitored for any ageing effects caused by fast neutron irradiation using the reactor vessel surveillance programme (RVSP). There are six capsules installed in each RPV that contain RPV beltline material and dosemeters to characterise the embrittlement of the material over the design life of the RPV. To date, the test data results indicate that a 60-year lifespan of the RPV is achievable. The unit 2 reactor pressure vessel head was replaced in 2022.

* **Pressuriser Heaters:** the pressuriser controls the primary coolant pressure. The lower end of the pressuriser is equipped with six banks of heaters that have a nominal rating of 1 440 kW and a surge line that connects to the Loop 1 hot leg of the reactor coolant system. The top end of the pressuriser is equipped with a spray system connected to the cold leg of the RCP. The upper end includes nozzles for connecting pressure relief valves and safety valves. Two of the heater banks (RCP 005 and 006 RS) have a qualified life of 40 years and will be replaced in support of LTO. The revalidation of the pressuriser heaters' qualified life supports the replacement of the heaters during LTO. The replacement of the pressuriser heaters is contained in the LTO IPP in Appendix A. The details of the design of the pressuriser are contained in SAR II-3.3.5 (*Pressurizer*) [178].

- Third barrier: containment building

  The containment buildings house the nuclear reactors, steam generators, reactor coolant pumps, and other primary system equipment and act as the third barrier to prevent the release of radioactive material to the environment during normal operation and design basis accidents. Furthermore, they protect the equipment inside from the external environment. The reinforced concrete containment buildings are post-tensioned and have a leak-tight steel liner. The design is similar to the containment of the EDF 900 MW fleet. The technical detail in terms of the design basis and the plant design interface is documented in SAR III-3.4 (*The Third Barrier – Containment*) [178].

  Due to significant chloride loading into the containment civil structure from the atmosphere at Koeberg that was not anticipated during the design stage, the external surfaces of the containment buildings have suffered from chloride ingress that causes rebar corrosion. Since the year 2000, various investigations, tests, and evaluations have been dedicated to the required recovery. The first was removing loose and spalled surface areas, followed by repairs. Several repair projects have been completed to date. However, it is clear that these efforts are temporary and not a permanent solution. An investigation by a group of international experts concluded that the only permanent solution was to protect the internal rebar and tendons through impressed cathodic protection. The investigation analysis is documented in JN465-NSE-ESKB-R-5704 (*Long-Term Repair Strategies for the Containment Buildings – Expert Panel Report*) [132]. A modification to provide such a system (based on the outcome of a mock-up) is in progress. Further detail regarding the ageing management of the containment structure is discussed in

§ 9.5.2.1. Ongoing improvements are being made to preserve and improve the condition of the containment buildings, and the technical evaluation of the TLAA, as documented in § 9.5.2.1, concluded that the containment structural integrity was ensured for the planned long-term operation.

9.4.2.1.2 Spent Fuel Facilities

The section below discusses the adequacy of the design of the major components related to the storage of spent fuel.

1. Refuelling water storage tanks

   The refuelling water storage tank design is adequate for long-term operation. These tanks were replaced on Unit 1 in 2018 (Outage 124) and Unit 2 in 2019 (Outage 224). The main reason for the plant change was to manage the atmospheric stress corrosion cracking in the tank shell, an ageing degradation mechanism. The new tanks are made of an improved material that is resistant to this degradation mechanism and, therefore, has a longer lifespan, thus maintaining the safety function for the entire period of LTO. The new tanks have also been designed and manufactured to withstand an earthquake more severe than the SSE, which further aids Koeberg's capability to manage design extension conditions involving earthquakes. The general design specifications and characteristics of the new tanks are described in SAR II-8.4.2.2 (*General Design*) [178].

2. Fuel building

   The plant design includes a fuel building containing a spent fuel pit area. This area is where the spent fuel is stored in racks covered by water. The design of the fuel storage is described in SAR II-1.9.4 (*Fuel Building*) [178]. The facility has been modified to increase pool storage capacity. It was initially designed in accordance with ANSI N 18.2-1973 and later modified according to ANSI/ANS 57.2-1983 to install super-high-density storage racks. The fuel storage facility is adequately designed to allow heat removal in all operational states and accident conditions and maintain the structural integrity of the fuel elements throughout the intended period of LTO and during the decommissioning phase of the plant.

   An assessment – 240-167231099 (*Assessment of the Spent Fuel Pool for Long Term Operation*) [80] – was performed to evaluate the need for an AMP for the neutron-absorbing material in the spent fuel pool. The assessment considered the design, materials, chemistry, operational experience, expected life and research associated with the neutron-absorbing material used in Koeberg's spent fuel pool. The assessment concluded that no specific AMP is required and the spent fuel pools can fulfil their neutron absorbing function for the period of LTO. However, the recommendations mentioned in the assessment need to be carried out for defence-in-depth and assurance purposes.

3. Fuel storage

   The fuel racks inside the spent fuel pool at Koeberg have been reaching maximum storage capacity, as the plant approaches the end of its operating licence. Additional spent fuel storage facilities are required to safely store the spent fuel produced during the LTO period. Therefore, Koeberg utilises dry storage fuel casks to safely store spent fuel that has been subjected to cooling in the spent fuel pool for a period of at least 10 years. The NNR approved the Transient Interim Storage Facility (TISF) licensing strategy 12010-D0001 [3], (Refer to NAPP11B033), which makes provision for additional dry storage casks for the planned Koeberg plant life extension from 2025 to 2045. This ensures that there will be sufficient storage until the centralised interim storage facility (CISF) is established by the National Radioactive Waste Management Institute (NRWDI).

   The licensing framework further explains that the Koeberg spent fuel management strategy, is to transport dry storage casks to the off-site CISF once it is established. Eskom is in the process of establishing the first storage pad of the TISF which is projected to be fully constructed and operational in 2024.

   The design bases for the dry storage casks are described in SAR II-8.1 (*Storage and Handling of Spent Fuel Shipping Casks)* [178].

4. Cask storage building (CSB)

   Initially, the cask storage building was relicensed to temporarily store a limited number of casks as part of the re-racking project. Further to this intervention, Koeberg intended to increase the number of storage casks and store more storage casks in the CSB for a longer period. This required additional analyses and assessments. In accordance with US design code standard ASCE 43-05, a new assessment concluded that the CSB building was not suitable for medium- to long-term storage of fuel casks. Modification 07147DPDRR0012 (*CSB for Fuel Storage Casks*) [2] has been raised to harden the structure against seismic hazards and other design-based external events. The CSB is capable of storing 11 casks for the full period of LTO. The modification to the CSB will improve its structural integrity to ensure that it will be qualified for the entire period of LTO.

5. Transient interim storage facility (TISF)

   A transient interim storage facility (TISF) is being constructed for the storage of additional casks. It will be utilised until the establishment of the central interim storage facility (CISF) by the government. Further detail regarding the use of the TISF for high-level waste is discussed in § 9.7.3.11.

9.4.2.1.3 Safety-related modifications

The safety improvement modifications listed below are based on the PSR plant design review [114].

1. Control room habitability

   The control room was built for the operators to monitor and operate the plant safely under normal operating conditions and maintain the plant safely under accident conditions. The control room provides protection to the control room occupants against high radiation levels resulting from an accident and releases of radioactive material with a dose limit not exceeding 50 mSv for the whole body. The PSR plant design review found this to be a deviation, as the dose limit could not be achieved based on the unfiltered in-leakage test performed in July 2021. Modifications to resolve the problem will be implemented in accordance with the LTO IIP. Several mitigating actions have been implemented to reduce the radiological consequences to the control room occupants in the event of an accident with radiological releases. These mitigating actions are listed in Appendix J (Suitability for Continued Operation) of the PSR global assessment report [115]. Subsequent to the discovery of the CRE criteria not being met, a safety justification J2021/0001 rev.1 (*Post SGR control room envelope justification for continued operation*) [131] was performed and concludes that the control room habitability criterion of 50 mSv is met when taking into account actual measured plant input parameters. Based on the safety justification the initial high risk identified during the PSR is no longer applicable. The safety justification was approved by the NNR in letter k29165N. Koeberg will continue to implement the planned modifications to improve the control room unfiltered in-leakage in order to reduce the dose to control room operators in case of an accident.

2. Nuclear component cooling system (RRI) pump room

   Appendix J of 331-608 (*KNPS 3rd Periodic Safety Review Global Assessment Report and Integrated Implementation Plan Report*) [115] states that the RRI pumps, which are located in the nuclear auxiliary building at the -6.7 m level, are protected against flooding at the highest postulated seawater level. The pumps are mounted on 0.5 m plinths. The design has considered the possibility of a flooding accident due to a maintenance or operational error. The PSR plant design review found flooding of the RRI pump room on two occasions. In the second flooding incident, the plant was close to losing all RRI pumps, and therefore the incident represents itself as a common cause of failure. Mitigating actions have been implemented to prevent such an incident and mitigate its consequences. Furthermore, to prevent the common cause failure of both trains, modification 03030C will be implemented. The modification is included in the PSR integrated implementation plan. The PSR plant design review, therefore, concludes that a timely resolution of this deviation will improve safe operation during LTO.

9.4.3 Design Basis for the Facility

Koeberg is a Framatome-designed plant with nuclear safety design criteria based on the ANSI N18.2−1973 code. The design basis of Koeberg considers the general principal design criteria for nuclear power plants, 10 CFR 50 Appendix A, to ensure that the operation of the plant is inherently safe throughout its operating life. The SAR I-4.3.2.3 (*Principles Applicable to the NSSS* *and Safety Systems – Analysis of the US NRC General Design Criteria*) [178] discusses how the Koeberg design meets the 10 CFR 50 Appendix A criteria. Koeberg complies with all the criteria.

The Koeberg design basis conforms to modern international codes and standards that ensure a high degree of confidence for the safe operation of the plant and the mitigation of potential events that can jeopardise the safety of the plant.

The PSR plant design review concluded that, although deviations with low significance were raised concerning several codes and standard shortfalls for the civil, structural, and mechanical design aspects of the plant, most plant designs conform to modern local and international standards.

The design basis has a range of conditions and events considered in the design of SSCs according to established criteria. The plant can withstand them without exceeding any limits. Where new plant design modifications are made, the design process allows for reconciliation analysis between new and old codes to ensure correct plant interfacing between old and new SSCs.

Koeberg considers the ageing effects of SSCs important to safety under design basis conditions, including transient conditions, and has postulated initiating event conditions in the specifications for equipment qualification programmes during all normal and accident conditions applicable to the equipment.

The SSCs important to safety at Koeberg are appropriately designed and configured to meet the requirements for the safe operation of the plant and the prevention and mitigation of events that can jeopardise safety. The design basis for all SSCs important to safety is systematically justified and defined in the SAR and is used as the basis for continuous safe operation.

KBA0022OTS0000001 (*Operating Technical Specifications (OTS)*) [163] ensures that the plant is safely operated in accordance with the normal operating design limits. The role and purpose of the OTS are to define the normal operating limits necessary to remain within the reactor design assumptions, define the operability requirements of safety functions, and prescribe the actions required if normal operating limits are exceeded or a required safety function is inoperable. Accident analyses were performed for Condition II, III, and IV events to demonstrate the ability to safely shut down the plant in the event of any design-based accidents. Design basis accident studies are described in SAR III-4.4.2 (*Rationale for Accidents Analysed*) [178].

Koeberg performs probabilistic safety assessments (PSAs) to assess the off-site radiological risk due to accidental releases of radioactive materials and ensure that it is within the regulatory criteria specified in RD-0024. The PSA results are documented in PSA-R-T19-01 (*Risk Assessment Report (RAR)*) [225] to demonstrate compliance with the regulatory requirements of RD-0024. The RAR is a design basis document and forms part of the safety case for continued safe operation. The RAR concludes that Koeberg has operated safely since commissioning based on the historical review data. The PSA model will be periodically updated to consider operating experience to ensure compliance with risk limits throughout the LTO period.

It is ensured that plant SSCs important to safety have appropriate design characteristics and are arranged and segregated in such a way as to meet the requirements for plant safety and performance. The plant SSCs are classified in accordance with 240-89294359 (*Nuclear Safety, Seismic, Environmental, Quality, Importance and Management System Level Classification Standard*) [84] to establish the SSCs that are important to safety. This classification allows appropriate management, maintenance, and testing to provide confidence that they will perform their safety function when required. The Eskom classification standard is in line with the classification requirements of the applicable international codes and standards.

Safety-related electrical components of the plant are designed and manufactured in accordance with the Institute of Electrical and Electronics Engineers (IEEE) rules and criteria. Safety-related mechanical components of the plant are designed and manufactured in accordance with the American Society of Mechanical Engineers (ASME) code. These codes provide the rules for classifying components based on the nuclear safety function of the component. These rules are described in 240-89294359 (*Nuclear Safety, Seismic, Environmental, Quality, Importance and Management System Level Classification Standard*), which is utilised when new components are added to the design of the plant through modifications. The rules and requirements are documented in SAR II-1.2 (*Classification of Equipment, Structures and Systems according to Nuclear Safety, Seismic, Environmental, Quality and Importance Levels*) [178].

The design and construction of the spent fuel storage facility are based on the ANSI 57.2-1983. This design code was used when the spent fuel pool facility was modified to install super-high-density storage racks to accommodate more than 10 years' worth of spent fuel storage (see § 9.4.2.1.2).

The PSR assessment concluded that the design basis of the existing plant was adequate for LTO. However, to ensure continuous improvement during the LTO period, safety improvements associated with codes and standards are included in the PSR IIP.

9.4.3.1 Reassessment of the Design Basis

The LTO assessments have not identified any need for design-based reassessment. Following the Fukushima Daiichi accident, a safety reassessment (EE-SRA) to evaluate the response of the plant against DECs was performed. Based on the outcomes of the safety reassessment, mitigating actions were identified and accepted by the NNR. The ongoing seismic studies as mentioned in § 9.4.6 are in progress and will be included in the DSSR when completed. Any potential design basis reassessment will be determined on the conclusion of the DSSR studies and the PSR IIP safety improvements related to the design extension conditions analysis, H_2 explosion studies, etc. At present, as discussed in § 9.4.2, the design basis remains unchanged and adequate for safe LTO.

9.4.4 Application of Defence in Depth in Plant Design

The prevention and limitation of accidents are based on the concept of defence in depth, first developed by the United States (US) Nuclear Regulatory Commission (NRC) in document WASH-

1250 (1973). This calls for the plant design to have adequate defence in depth through a combination of several layers of protection.

SAR I-4.3.2.2.1 (*Defence in Depth*) [178] discusses the application of the defence-in-depth concept. The approach is according to standard ANSI N18.2-1973 and according to US NRC document WASH-1250 (1973), which requires that three levels of defence be covered. Where ANSI-18.2 was inadequately detailed, ANSI 51.1-1983 has been used in modifications and classifications, except for the spent fuel pool storage, which was designed based on ANSI 57.2-1983. The levels of defence in depth are accident prevention through conservative design and quality of fabrication (Level 1), provision of protection systems to stop the development of an accident (Level 2) and control of accidents to limit radiological releases and prevent escalation to core melt conditions (Level 3). The SAR also notes that "In practice, a fourth level of defence now exists with emergency operating procedures for coping with beyond-design-basis accidents and the emergency plan" (SAR I-4.3.2.2.1 (*Defence in Depth*)).

The approach to the application of defence in depth does not fully align with current international relevant good practice guidance provided in the later IAEA INSAG-10 standard, which introduces two additional levels, that is, to control accidents with significant core melt to limit off-site releases (Level 4) and mitigation of the radiological consequences of radioactive releases that can potentially result from accidents (Level 5).

The Koeberg SAR, which focuses on Levels 1, 2, and 3, is not fully aligned with the approach documented in IAEA INSAG-10; however, evidence of elements of Levels 4 and 5 does exist at Koeberg such as the severe accident guidelines and associated design provisions to support the severe accident management guidelines (SAMGs) and a comprehensive emergency plan.

The PSR plant design review captured this shortfall as a deviation with a safety significance of low. The safety improvement action for this deviation entails Koeberg's achievement of full alignment with current international relevant good practice guidance for the defence-in-depth concept, which gives a refined structure of the five levels of defence in depth. It is noted that this shortfall does not preclude safe plant operation or safe LTO. In accordance with RG-0028, the safety improvements will be completed within a reasonable time before the next PSR, following the NNR's concurrence with the PSR IIP. The adequacy of the design provisions for DiD is discussed in detail in Appendix D.

9.4.5 Fulfilment of the Fundamental Safety Functions

The Koeberg requirements as documented in the SAR are broadly in line with the fundamental safety functions as specified in international relevant good practice guidance provided in applicable IAEA SSR-2/1 and IAEA General Safety Regulations (GSR) Part 4 standards, that is:

- control of reactivity;
- removal of heat from the reactor and fuel storage; and

- confinement of radioactive material, shielding against radiation, control of planned radioactive releases, and limitation of accidental radioactive releases.

SAR I-4.3.2.5.1 (*Protection Against External Hazards*) states that the fundamental requirements of the plant are to:

- preserve primary circuit integrity;
- shut down the reactor and remove the residual energy; and
- limit the release of radioactive substances at the site boundary to an acceptable value.

SAR I-4.3.2.3.21 (*Protection System Requirements for Reactivity Control Malfunctions*) recognises the importance of the reactivity control systems to ensure that specified acceptable fuel design limits are not exceeded due to anticipated operational occurrences.

Several SAR chapters deal with the SSCs that fulfil the fundamental safety functions stated above. The information below is a general summary from the SAR, noting that further detail on the individual systems is provided in the SAR and associated DSE (Dossier de Système Élémentaire) chapters.

The PSR plant design review concluded that the plant design is adequate for the fulfilment of the FSFs. The global assessment concluded that some deviations identified during the review phase of the PSR have an impact on the FSFs, and the safety improvements to address these are contained in the PSR IIP [115]. Further details on the findings of the global assessment relating to the fulfilment of the FSFs are discussed below in § 9.4.5.4.

9.4.5.1 Control of Reactivity

Concerning the control of reactivity, as noted above, SAR I-4.3.2.3.21 states the need for the protection system to, firstly, be designed to automatically initiate the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded due to anticipated operational occurrences and to, secondly, sense accident conditions to initiate the operation of SSCs important to safety.

The protection system is designed to limit reactivity transients so that fuel design limits are not exceeded. To achieve this, a fully automatic reactor protection system (RPR) with appropriate redundant channels is provided to cope with transients where insufficient time is available for manual corrective action. The design basis for the protection system is in accordance with IEEE Standard 279-1971.

The reactor protection system automatically initiates a reactor trip when any variable monitored by the system or combination of monitored variables exceeds the normal operating range. Set points are designed to provide an envelope of safe operating conditions with an adequate margin for uncertainties to ensure that fuel design limits are not exceeded.

Core reactivity is controlled by:

- adjusting the concentration of boric acid dissolved in the coolant;
- placing burnable poison rods in selected fuel assembly guide tubes; and
- moving the rod cluster control assembly (RCCA) into the core.

9.4.5.2 Removal of Heat from the Reactor and Fuel Storage Pool

Concerning the removal of heat from the reactor, SAR I-4.3.2.3.30 (*Residual Heat Removal*) [178] states the need for a system to remove residual heat from the reactor. The system safety function is to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

The steam generator main feedwater system (ARE) allows the residual heat produced in the core and heat from other sources after subcriticality during reactor shutdown to be removed without exceeding the design limits for the fuel and the reactor coolant system pressure envelope. Normally, the system is used during the first phase of cooldown but is dependent on the availability of the turbine bypass system (GCT) and the condenser.

The auxiliary feedwater system (ASG) can be used as a backup to the main feedwater system in conjunction with the GCT atmospheric relief system to remove residual heat from the core.

One or more reactor coolant pumps ensure circulation of reactor coolant in the core if the off-site power supply is available and, otherwise, by natural circulation.

At lower pressures, reactor coolant circulation and core cooling are ensured by two pumps and two heat exchangers of the residual heat removal system (RRA), which are themselves cooled by the component cooling system (RRI), which, in turn, is cooled by the essential service water system (SEC).

Both RRA pumps and heat exchangers and RRI and SEC trains would normally be in service during cooldown.

The safety injection system (RIS) provides an emergency core cooling system to cope with any loss-of-coolant accident (LOCA) in the primary coolant system.

SAR I-4.3.2.3.34 (*Containment Heat Removal*) [178] states the need for a system to remove heat from the reactor containment. The system safety function is to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

Concerning the removal of heat from the fuel store, SAR I-4.3.2.3.52 (*Fuel Storage and Handling, and Radioactivity Control)* [178] states the need for a fuel storage system with a reliable residual heat removal capability, having a functional capability for increased cooling capacity, component testing, and maintenance during various modes of plant operation.

Irradiated fuel is stored underwater in racks in the spent fuel pit. Spent fuel pit water is circulated through the spent fuel pit cooling system (PTR) to maintain water temperature, purity, and clarity according to specifications.

The installation is designed to ensure the safe containment of radioactive substances. Piping outlets from the spent fuel pit are at a level that ensures that stored elements remain covered by water should the PTR cooling system develop a leak. The system provides reliable, redundant heat removal capabilities with backup electrical supplies from the other unit.

The installation is designed to prevent a significant reduction in fuel storage coolant inventory under accident conditions by the design of the spent fuel pit to seismic Class 1 and provision of a demineralised water distribution system (SED) and firefighting water distribution system (JPD) supplies as backup to PTR.

The fuel storage and handling systems are described in more detail in SAR II-8 (*Storage and Handling of Spent Fuel Shipping Casks*) [178].

9.4.5.3 Confinement of Radioactive Material

Concerning the confinement of radioactive material, shielding against radiation and control of planned radioactive releases, and limitation of accidental radioactive releases, a steel-lined, pre-stressed concrete containment encloses the reactor coolant system (SAR I-4.3.2.3.12 (*Containment Design*) [178]). With a few exceptions, all penetrations through the containment enter penetration rooms in the nuclear auxiliary and fuel buildings, which are separately ventilated to collect and process any penetration leakage. The containment isolation will limit leakage by providing a leak-tight barrier against the spread of radioactivity that may be released into the containment atmosphere in the unlikely event of a serious accident. Other systems provided to lessen the amounts of radioactivity that may leak from the containment to the environment are the safety injection system (RIS) and the containment spray system (EAS), limiting the pressure and temperature inside the containment to values below the design conditions for all postulated accidents.

SAR I-4.3.2.3.28 (*Inspection of Reactor Coolant Pressure Boundary*) [178] indicates that components of the reactor coolant pressure boundary are designed to permit periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity and are subject to an appropriate material surveillance programme for the reactor pressure vessel. The reactor coolant pressure boundary is periodically inspected, which complements the various leakage collection systems in assessing the integrity of the pressure boundary components.

SAR I-4.3.2.3.48 (*Reactor Coolant Pressure Boundary Penetrating Containment*) [178] notes that each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment is provided with containment isolation valves. The specifications of this provision are aligned with 10 CFR 50.2; the reactor pressure boundary extends to the outermost containment isolation valves in lines that are connected to the primary system. A detailed description of the isolation arrangement of each piping penetration and a comparison of the arrangement with the criterion with exceptions are contained in SAR II-4.2.4.

SAR I-4.3.2.3.51 (*Control of Releases of Radioactive Materials to the Environment*) [178] details that the design includes the means to control the release of radioactive materials in gaseous and liquid effluents and manage radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Waste treatment systems have been incorporated into the facility design to process or retain radioactive wastes from normal operation and anticipated operational occurrences. Controls and monitors capable of closing discharge isolation valves are provided to ensure that releases are in accordance with the NNR's LD-1020 (*Radiation Dose Limitation at Koeberg Nuclear Power Station*) [284].

SAR I-4.3.2.3.55 (*Monitoring Radioactivity Releases*) [178] details that, in addition to measurement of radioactivity by a sampling of the effluents before release, devices are provided for continuous measurement of the activity of potentially contaminated releases. In addition, samples are taken in the power station area to monitor the ambient radioactivity: surface water, groundwater, food pre-products, etc.

The containment spray system (EAS) is designed to rapidly decrease containment temperature and pressure following a loss-of-coolant accident. The system consists of two redundant trains, each capable of ensuring 100% of the safety function.

SSC classification-related processes and procedures, thus, allow the components for the systems mentioned above to be classified based on their role in achieving the fundamental safety functions.

331-195 (*Koeberg Accident Analysis Manual*) [102] describes the process for performing accident analysis. An objective of the analysis is to verify the adequacy of the plant design (that is, the SSCs and the barriers incorporated into the design) and the capacity of the safety systems to fulfil the safety functions required of them.

9.4.5.4 Condition of the FSFs

The PSR global assessment identified 18 deviations that could affect core cooling and the confinement of radioactive material FSFs. One deviation was graded as "high" and is linked to the control room not being sufficiently protected against the ingress of radioactive material. Another deviation was graded as "medium" and is linked to the modifications identified to mitigate the effects of external hazards beyond the design basis and DEC-A conditions. The remaining 16 deviations were all graded as "low". The impact analysis of the deviations for the FSFs concluded that no significant cumulative effect exists, and no global issue was raised. The PSR plant design review determined the current status of plant safety to be sufficient for safe operation and LTO provided the safety improvements identified in Appendix A are implemented. Therefore, the FSFs are not challenged and are deemed adequate for LTO.

9.4.6 Design Safety Margins and Safety Analyses

9.4.6.1 Design Safety Margins

LTO for Koeberg is supported by the availability of sufficient safety margins incorporated into its plant design.

WANO GP ATL-11-005 (*Excellence in the Design and Management of Design and Operating Margins*) [299] guides utilities in finding, evaluating, prioritising, and resolving safety margin concerns. It states that conservatisms incorporated into system design and operational limits (that is, the design and operating margins) ensure that operators and plant systems have sufficient flexibility to accommodate routine activities and respond to anticipated transients and accident scenarios effectively. Careful configuration control, evaluation of changes, and equipment condition monitoring are noted as necessary to maintain acceptable levels of design and operating margins.

IAEA GSR, Part 4, Requirement 13 (*Safety Assessment for Facilities and Activities*) [246] requires that the plant design ensure adequate safety margins. There is a wide margin to failure of any SSC important to safety for any anticipated operational occurrences or possible accident conditions. Considering this requirement, 36-197 (*Koeberg Licensing Basis Manual (KLBM)*) [125] indicates that plant design activities are conducted to encompass prudent safety margins within the design. This is as required by WANO GP ATL-11-005 (*Excellence in the Design and Management of Design and Operating Margins*) [299]. For example, SAR I-4.3.2.3.11 (*Reactor Coolant System Design*) [178] is specific to reactor coolant system (RCP) design, noting that the associated SSCs achieve a large margin of safety by using proven materials and design codes and proven fabrication techniques, non-destructive testing, and system leakage testing of assembled components.

Section 3.11 of 331-195 (*Koeberg Accident Analysis Manual*) [102] concerns margin evaluation. It presents the margin model provided by WANO GP ATL-11-005 and details that this provides the basis for the principles of margins as accepted by Koeberg. Design margin accounts for design assumptions used in calculations, equipment tolerances such as structural component dimensions, instrumentation tolerances, calculation round-off, and allowance for degraded equipment performance. The accident analyses must provide an engineering document that includes a summary of the margins between the calculated results from the limiting case and the corresponding acceptance criteria.

The design of Koeberg has ensured that sufficient provision for defence in depth exists. Safety margins are ensured by applying the general design principles prescribed in 10 CFR 50 Appendix A in the design of the plant, such as defence in depth, single failure criterion, redundancy, and diversity. The design of the safeguard systems caters for the possibility of the single failure of passive and active components important to safety. The single failure criterion also applies to the Koeberg design of the safety-related fluid systems provided to mitigate the consequences of Condition III and Condition IV events as described in ANSI N18.2-1973. The Koeberg design caters for the redundancy of safety systems by incorporating an independent duplicate of each safety system, known as Train A and Train B, for each unit. The design, furthermore, caters for diversity in some cases such as the ASG pumps, with ASG 001 and 002 PO being electric-motor-driven and ASG 003 PO steam-turbine-driven pumps performing the same function.

Operationally, throughout the life of the plant, safety margins are maintained through the following:

- Adequate qualification of SSCs important to safety
- Adequate execution of ageing-management-related programmes for SSCs important to safety
- Operation of the plant within the operating limits and conditions
- Use of the technical specifications
- Appropriate plant upgrades where margins can no longer be maintained
- Additionally, the loading on the plant is monitored and a transient report is maintained. To adequately monitor the loading on the plant, the SSCs with critical loading limitations have been identified, and the loading is monitored.

All of these aspects mentioned above related to maintaining safety margins were reviewed in the PSR actual condition of SSCs assessment [114] and found to support entry into LTO. Where deviations were identified, the safety improvements to address these will be implemented in accordance with the LTO IIP. Additionally, the PSR actual condition of SSCs assessment [114] concluded that there were processes to ensure that the safety margins were known. Where margins were degraded, adequate programmes were embedded to ensure that the necessary component replacements or refurbishments were made to improve safety margins.

9.4.6.2 Safety Analyses

The design of the facility is informed by extensive and comprehensive safety analyses that have confirmed that the facility can operate for an additional 20 years. This section discusses the safety analysis aspects, namely, deterministic safety analysis, probabilistic safety assessment, and hazard analysis, related to the design of the facility.

- Deterministic safety analysis

  The deterministic safety analysis performed considered a comprehensive list of postulated initiating events to assess the integrity of the barriers during the life of the plant. These postulated accidents do not yield the same consequences or the probability of occurrence. Therefore, they are classified into different categories (Condition I to Condition IV) based on their frequency and radiological effects in accordance with the ANSI N18.2-1973 standard. SAR section III-4.3.1.1 (*Classification of Operating Conditions*) [178] describes possible events for each condition as mentioned in ANSI N18.2-1973. The system DSE (Dossier de Système Élémentaire) manual, Chapter 7, describes the safety analysis of the nuclear steam supply systems (NSSSs) in detail.

As discussed in § 9.1.1.5, the results of the PSR deterministic safety analysis review indicated that the facility has comprehensive DSA and that:

* the DSA for design basis accidents is aligned with national and international standards; and
* the DSA for design extension conditions was improved for alignment with national and international standards.

Although the plant is not yet fully aligned with the international requirements related to DEC, the DEC related deviations do not have a significant impact on the existing ability of the plant to prevent and/or mitigate DBA, DEC-A, or DEC-B accidents.

Probabilistic safety assessment

The probabilistic safety assessment (PSA) adequately complements the DSA, and it is a comprehensive and structured analytical tool for identifying accident scenarios and deriving numerical estimates of risks of undesirable consequences concerning the operation of the facility and its associated plant vulnerabilities and, as such, is suitable for making risk-informed decisions relating to the design of the plant.

Through the PSR PSA review, it was demonstrated that the PSA is adequate for risk-informed applications to support engineering, operations, and nuclear safety decision-making within the current scope of the model and to demonstrate licence compliance. PSA is used extensively at Koeberg to support safety management and decision-making through the safety evaluation, non-conformance, and qualitative risk processes. As it relates to the design aspects, PSA is used extensively to complement DSA; for example, it is used in the definition of safety classification of SSCs important to safety and to determine the impact of changes to operating technical specifications, procedures, etc.

The PSR demonstrated that the PSA remains valid as a representative model of the plant, that the current plant design and operating features have been modelled to a sufficient level of detail, and that the results are sufficiently well-balanced for all postulated initiating events (PIEs) and operating states.

The ASME peer reviews concluded that the PSA generally meets Capability Category II, and therefore, the existing scope, capability, and application of the PSA are sufficient for licence compliance and risk applications. The review demonstrated that the existing PSA is sufficient to support the PSR GA.

The review identified 23 deviations: 21 were graded to have “low” safety significance and two “drop” safety significance. The deviations do not impede the continued safe plant operation, and the identified safety improvement actions for the deviations need to be implemented and prioritised in accordance with the deviation safety significance grading.

A deviation related to the update of the emergency plan technical basis (EPTB). The EPTB needed to be reassessed to consider the impact of the significant safety improvements made at the facility in recent years, the pursuit of LTO, the impact of new regulatory guidance on the EP, and new international EP requirements published by the IAEA in recent years incorporating OE and lessons learnt from the accident at Fukushima Daiichi nuclear power plant (NPP). The EPTB update has been completed and submitted to NNR.

Regarding external hazard PSA, Eskom recently committed itself to a strategy with the Regulator to undertake a seismic re-evaluation of the plant, of which reassessment of the seismic PSA was the cornerstone. As such, it was imperative that the seismic PSA be developed in the near term. This is not a prerequisite for LTO.

This review confirmed that the overall PSA results demonstrate that the risk associated with the operation of the facility is well within the principal safety criteria (risk limits) specified in RD-0024 (*Requirements on Risk Assessment and Compliance with Principal Safety Criteria for Nuclear Installations*) [289] and, therefore, supports safe continued operation. The deviations do not impede the continued safe plant operation, and the identified safety improvement actions for the deviations have been included in the PSR IIP.

- Hazard analysis

Hazard analysis was performed to ensure that the plant was adequately designed against internal and external hazards. During the PSR hazard analysis review, the robustness of the plant against internal and external hazards was assessed. The assessment concluded that the plant is mainly robust against internal hazards, with some vulnerabilities identified related to external hazards. Six deviations with a “medium” safety significance grading were raised mainly regarding robustness of the plant against external hazards. The medium deviations relate to the updating of the hazard-related information in the site safety report, the absence of the hydrogen study, and the EE-SRA modifications raised that had not been implemented. Koeberg is cognisant of the risk posed by these external hazards and provisions have been made to mitigate these hazards. Improvement plans for these deviations are incorporated in the PSR IIP. The aspects of the site-specific external hazards § 9.3 were assessed, and the plant was found to be robust.

Document 240-160677773 (*Koeberg Seismic Re-Evaluation Strategy*) [61] documents the strategy for demonstrating that, from a seismic hazard perspective, the plant is adequately safe from seismic activity. An interim seismic evaluation was performed to provide confidence in the seismic robustness and safety of the plant. The interim evaluation was based on the approach developed by EPRI, referred to as the expedited seismic evaluation process (ESEP), which the US NRC utilises. The objective of the ESEP is to demonstrate seismic margin through a review of a limited, but justified, scope of equipment that can be relied on for the safe shutdown of the plant following a significant seismic event without affecting regulatory safety criteria.

The ESEP assumes the most credible scenario following a significant seismic event, that is, all alternating current (AC) power loss. It guides the selection of equipment required to prevent radiation impact on staff and the public. The equipment selection was done following the ESEP guidance and augmented with logic analysis and adaptation to the Koeberg plant design.

The robustness verification was done using the ESEP guidance by computing (or scaling) the original SSE to obtain a review-level earthquake (RLE). The ESEP process and evaluation are documented in 32-T-IPDK-002 (*Interim Seismic Evaluation for Koeberg NPS*) [91].

The ESEP interim seismic evaluation provided reasonable assurance that the Koeberg units are sufficiently robust to shut down safely and cope with a significant seismic event and loss of AC power when certain activities had been performed. These activities are mentioned in the conclusion of the 32-T-IPDK-002 [91], and these are included in the LTO Implementation Plan, Table A.2-3.

The lack of an up-to-date site safety report was deemed a shortcoming in the PSR hazard analysis review and hence a deviation with a "medium" safety significance was raised to capture this issue. In accordance with § 9.3, the DSSR is being updated to address the identified deviation. The outstanding SSHAC study will be completed and incorporated into the DSSR by 2024. The revised DSSR will be submitted to the NNR for approval, and all the design basis documents impacted by the changes in the hazards will be updated.

9.4.6.3 Control of Safety Analysis Changes

The safety analysis is continuously evaluated during changes to plant and general operating rules to ensure that there are no uncontrolled changes to the licensing and design basis. In accordance with the safety evaluation process procedure 240-143604773 (*Safety Screening and Evaluation Process – KAA-709*) [47], safety evaluations are performed to assess the impact of the proposed change on the existing safety analysis as described in the SAR. The above-mentioned safety evaluation process is based on the requirements of 10 CFR 50:59. It is utilised to ensure that plant changes are properly evaluated for their impact on the design basis and current licensing basis. This procedure was found to be adequate during the PSR review of procedures. If the safety evaluation finds any adverse safety impact, a safety justification is compiled to justify the implementation of the change or continued safe operation and regulatory approval is obtained.

9.4.6.4 Ageing Analyses in Design

Koeberg has created 240-132364298 (*Initial List of Time-Limited Ageing Analyses for Koeberg Nuclear Power Station*) [43], a list to capture all TLAAs for SSCs important to safety. This process is incorporated into the design process to track any changes associated with TLAAs. This list, developed in accordance with IAEA guide SSG-48, meets the requirements of the regulatory guide RG-0027. The TLAAs at Koeberg were verified by means of comparison with the IAEA SRS-82 IGALL list and were found to be comprehensive. All TLAAs identified will be revalidated or justified for 60-year plant life prior to LTO. Significant progress has already been made in this regard. The details of the outstanding analyses are documented in § 9.5.

9.4.7 Configuration Management of Design Basis Documents

Koeberg processes and procedures are in place to fulfil the requirements of a design management system and ensure that accurate information, consistent with the physical plant and operational characteristics, is available in a timely manner. This enables safe, well-informed decisions to be made. The processes are broadly aligned with international good practices and ensure that plant design changes are effectively controlled and are fit for purpose. There is a good correlation between the three pillars of configuration management (that is, design requirements, paper plant, and physical plant).

The Koeberg integrated management system is based on ISO 9001 and complies with RD-0034. The objective of the IMS is to ensure that Eskom nuclear installations are sited, designed, manufactured, constructed, operated, and decommissioned in accordance with national and Eskom policy, regulatory, and nuclear installation licence requirements and to provide the measures required to prevent or address non-conformance of products or unsafe processes. The manual 238-8 (*Nuclear Safety and Quality Management Manual*) [28] defines the IMS requirements to ensure that nuclear and radiological safety is appropriately considered for all activities that may be affected by safety throughout the life cycle of the plant. Furthermore, it describes the design management and configuration management requirements.

Koeberg has a design management system that ensures that all safety requirements established for the design of the plant are considered and implemented in all phases of the design process. The design management system ensures that the design process covers the quality of the overall design. The design management system also ensures that the control of the plant design and configuration is kept up to date and maintained throughout the life of the station.

Koeberg procedures fulfil the requirements of a configuration management system. The processes and procedures ensure a robust configuration management system. To ensure the configuration between the physical plant and the design documents, configuration management requirements associated with plant changes are documented in KAA-501 (*Project Management Process for Koeberg Nuclear Power Station Modifications*) [134].

The design documentation change process [120] defines the roles and responsibilities for making design plant changes. Document 331-86 (*Design Changes to Plant, Plant Structures or Operating Parameters*) [121] ensures that design changes to SSCs or changes to operating parameters are correctly compiled, updated, reviewed, and approved to ensure that all design configuration updates are met.

Document 240-143604773 (*Safety Evaluation Process*) [47] also forms part of the design change process to ensure that design changes are performed safely. The safety evaluation process defines the different levels of safety assessment (safety screening, safety evaluation, safety justification, and safety case) when making any design or plant changes.

The Koeberg configuration management system ensures that nuclear safety-related documentation and records management is within Eskom corporate and record management standards and policies. The Koeberg configuration management system has been developed in accordance with the guidance of the IAEA configuration management report IAEA-TECDOC-1335 (*Configuration in Nuclear Power Plants*) [270] and IAEA-SRS-65 (*Application of Configuration in Nuclear Power Plants)* [261]*.* Document 238-6 (*Standard for Nuclear Documentation and Records Management Requirements*) [27] establishes the nuclear documentation and records management requirements for Koeberg as defined in Document 238-8 (*Nuclear Safety and Quality Management Manual*) [28]. Document 331-3 (*Nuclear Engineering Documentation and Records Management Work Instruction*) [108] describes the activities and interfaces with the Nuclear Engineering documentation control centre to manage all nuclear engineering documents, including compilation, review, authorisation, publication, withdrawal, and archiving. The nuclear operating unit (NOU) configuration management process manual defines the high-level configuration process and requirements for all technical documents for the NOU.

An assessment of the sufficiency of the design management system and configuration management system in line with industry requirements and practices was performed during the PSR plant design assessment [114]. The assessment demonstrated that the processes and procedures are aligned with international good practices and provide an effective design and configuration management system. No regulatory compliance gaps were found. The PSR plant design assessment concluded that no significant plant configuration management issues were found [114].

9.4.8 Design Documents

Koeberg's processes and procedures allow for design documents to be representative of the physical plant. Koeberg has sufficient design documents, and where there are no documents available, the strategy to obtain design-based documentation is through the establishment of long-term contractual agreements with some original equipment manufacturers and other utilities. The plant design documents play a significant role in ensuring that the plant is operated safely within its operating limits. The document system ensures that the plant SSCs are correctly maintained for the plant to be efficient and provides confidence that the plant is safe. The PSR plant design assessment found no significant findings; however, a deviation with low significance was identified regarding the need to update affected documents resulting from modifications in a timely manner [114]. Therefore, it could be concluded that the design documentation is deemed adequate for LTO.

The main documents describing the plant design are as follows:

- Safety analysis report (SAR)

The Koeberg SAR documents the design basis in terms of the design principles, assumptions, rationale, criteria, and considerations used for the calculations and decisions made for the design. The SAR is divided into three parts. Part I contains the general layout of the plant, describing the major systems and buildings. Part II contains information on the SSCs necessary to maintain Koeberg in a safe condition during all operating states. Part III contains information on quality assurance during operation, the Koeberg nuclear emergency plan and its technical bases, and design basis accident analyses and radiological impact. The SAR is a live document updated regularly with plant changes or new studies. All SAR changes require NNR approval, and the process for updating the SAR is documented in procedure 240-119744497 (*Control of the Safety Analysis Report*) [37].

- Protection design files

  The design protection and safeguard systems analyses are presented over six files focusing on different accidents or SSCs. The protection design files summarise the design protection and safeguard systems analyses. Updating these files is a requirement of the design change process 331-86 (*Design Change Process*) [121] when there is a design basis or accident analysis change.

- Site safety report (SSR)

  The SSR provides the site characteristics supporting the design basis of the plant, will be subject to continuous safety reassessment, and will fulfil the NNR requirements. As noted in § 7.0, the DSSR will supersede the existing KSSR once the ongoing SSHAC study has been incorporated and following approval by the NNR. The DSSR lists and evaluates the natural and human-made hazards that can affect the safety of the plant and the cumulative radiological impact on the public and the environment. An update to the DSSR is initiated when regulatory requirements change, international good practices such as the IAEA safety guides are improved, and changes to the current environment manifest as a result of the 10-yearly periodic safety review.

- Operating technical specifications (OTS)

  KBA0022OTS0000001 (*Operating Technical Specification (OTS)*) [163] defines the normal operating limits necessary to remain within the reactor design assumptions, defines the operability requirements of safety functions, and prescribes the actions required if normal operating limits are exceeded or a required safety function is inoperable. The OTS is mainly used as an instruction manual by the plant operators to safely guide and operate the plant within operational limits. The bases for the operational limits are documented in the OTS justification manual, KBA022OTSJUSTIF1 (Chapter 1), KBA022OTSJUSTIF2 (Chapter 2), and KBA022OTSJUSTIF3 (Chapter 3).

- Chemistry specification

The chemistry specification manual KBA0022CHEMSPEC00 (*Chemistry Specification Manual*) [161] defines the chemistry and radiochemistry limits to ensure that the plant is safely operated within these limits. The chemistry technicians use this manual in conjunction with operators by referencing the OTS operational limitations. The bases for the chemistry specification limits are documented in KBA0022CHEMJUSTIF1 and KBACHEMJUSTIF2. The OTS and chemistry specification manuals are updated when a change is required using procedure 240-149081050 (*Control of the Operating Technical Specification*) [51].

- Safety-Related Surveillance Manual (SRSM)

  KBA0022SRSM00000 (*Safety-Related Surveillance Manual (SRSM)*) [164] defines the periodic testing programme, specifies the scope of the periodic testing requirements, describes the testing principles, and describes the form and manner in which periodic testing is performed. The objective of the SRSM is to ensure that the designed level of safety is maintained by periodically testing all SSCs important to safety, which provides a sufficient degree of confidence that the SSCs will perform their safety function when required. The testing methodology for each SSC important to safety is described in a test rule document for each system. The basis for the testing criteria is documented in an exhaustive analysis document for each system. The maintenance technicians use the SRSM to perform the testing in conjunction with the operators by referencing the OTS operational limitations. The SRSM is updated when an SRSM change is required or requested due to a design modification when the testing methodology changes or a benchmarking exercise that calls for changes. The process of updating the SRSM, test rule documents, and exhaustive analyses is documented in procedure 240-143370657 (*SRSM Change Process*) [46].

- Dossier de Système Élémentaire (DSE) (Design Manual)

  Koeberg has various design documents that describe the physical and technical design aspects of the plant. The DSE (Dossier de Système Élémentaire) manuals describe the design basis for each plant system. The DSE manual is structured into 14 chapters. Each system manual describes the system function, the design basis, and the function of each system component in detail. The DSE provides lists of various mechanical, electrical, and instrumentation components that make up the system, along with their component specifications. The DSE manual consists of various diagrams, such as flow diagrams, control logic diagrams, instrumentation diagrams, electrical wiring diagrams, and computer diagrams to support understanding of the functionality of the system. The DSE also provides important control room information to the operator to help operate the plant safely. In addition to the DSE manuals, Koeberg has other documents that provide more detailed information that is discipline-specific to the plant. The electrical board outage sheets and the electrical board supply feeder diagrams provide information regarding the power supplies and circuit breakers to which the components are connected, which help with maintenance and isolation. The welding isometric drawings provide information regarding welds and pipe layouts. The control room alarm documents provide information on, and diagrams of, each alarm panel in the control room. These documents are controlled and updated when modifications to the physical plant are made.

9.5 Ageing Management for Long-Term Operation

The primary objective of ageing management (AM) is to ensure that the effects of plant ageing will be adequately managed and that the asset is in a fit-for-service condition (integrity, safety, reliability) while extending its remaining life in the most reliable, safe, and cost-effective manner.

The section discusses the current status of AM at Koeberg as concluded by the ageing management evaluation (AME) performed in the SALTO project, as well as the programmatic arrangements for AM during LTO. A comprehensive AM assessment was performed through the SALTO project, which commenced before PSR. The ageing management evaluation included all SSCs important to safety. However, during the PSR ageing review, the progress made in the SALTO project was considered in the ageing review (in line with the PSR cut-off timelines) and the grading of the associated deviations.

Additionally, the section discusses specific SSCs considered to be a risk in terms of ageing management, namely the containment, aseismic bearings, switchboards and cables. Their risks and how these are mitigated to ensure that the SSCs can continue to reliably perform their safety functions are discussed.

The section, furthermore, addresses the ageing management of the SSCs that do not meet the AM scoping criteria defined in RG-0027 but are important to support the effectiveness of the license binding programmes. The assessment performed demonstrated that effective ageing management practices and processes to prevent the adverse effects of ageing from affecting the reliability of the plant equipment during the period of LTO exist at Koeberg.

9.5.1 Ageing Management Assessments

The section aims to summarise the results of the ageing management assessments and to provide a position on the adequacy of the physical ageing management programmes.

Koeberg has implemented an ageing management approach to ensure that degradation mechanisms and ageing effects are managed in a manner that SSCs deemed important to safety will continue to fulfil their intended design function and to ensure that the intended safety functions of these SSCs will be maintained with sufficient design safety margins for the LTO period. The results of the ageing management review support the continued operation of the plant, provided that the LTO IIP is implemented timeously to ensure safe LTO.

An ageing management evaluation [57] determined that the effects of ageing are adequately managed. The ageing management evaluation process included an assessment of the current physical condition of the SSCs, the identification of ageing degradation mechanisms, the review and validation of the appropriate plant programmes and processes for ageing management, and the identification of all TLAAs.

To meet the safety objectives mentioned above, sections 6 and 7 of the interim regulatory guide RG-0027 prescribe the activities to ensure effective ageing management of SSCs important to safety. The ageing management activities for LTO at Koeberg are progressing to ensure that the plant fully complies with the interim regulatory guide RG-0027 [294].

To determine the current condition of the SSCs and the ageing management status, the required ageing management evaluations were conducted under the SALTO project and were carried out in accordance with RG-0027 and the IAEA-SSG-48 [265] guidelines. The outcome of the SALTO assessment 240-156945472 (*SALTO Ageing Management Assessment Report (Interim)*) [57] indicated that the plant is suitable for an additional 20 years of safe operation. The SALTO assessment concluded that "*The assessment results and recommendations have been reviewed, verified, and actioned to meet the regulatory expectations and assure safe operation into LTO. This result will form part of the application to obtain the required nuclear installation licence variation before LTO.".* The ageing management activities performed are discussed below. Subsequent IAEA SALTO support missions have been held to provide KNPS with further suggestions and recommendations in improving preparation for safe LTO which have been incorporated into KNPS LTO preparation plans.

9.5.1.1 Scope Setting

The scope-setting process aimed to identify all SSCs important to safety subjected to ageing management. The scope setting process (which included both the scoping and screening tasks) for the important-to-safety SCCs was performed using 240-125839632 (*Koeberg Long-Term Operating (LTO) Scoping Methodology*) [40]. A list of SSCs important to safety was developed and maintained and kept current through the classification process as described in the classification guide 331-93 (*Guide for Classification of Plant Components, Structures, Parts, Services and Software*) [123].

9.5.1.2 Ageing Management Review

The ageing management review (AMR) [57] for in-scope SSCs was performed to determine whether ageing management processes and activities at Koeberg were comprehensive and to ensure that ageing effects were effectively managed so that the intended function of the SSCs would be maintained for the LTO period. The ageing management evaluation procedure 240-125122792 (*Koeberg Safety Aspects of Long-Term Operation (SALTO) Ageing Management Evaluation Process and Revalidation of the Time-Limited Ageing Analyses*) [39] explains the methodology for performing the AMR.

9.5.1.3 Ageing Management Programme (AMP) Review

An ageing management programme review [57] was performed to determine the adequacy and effectiveness of the existing ageing management programmes and plant processes for all the in-scope SSCs. The effectiveness and adequacy of the existing AMPs were determined by verifying the consistency of the AMPs using the nine attributes of an effective AMP, as indicated in Annexure A of RG-0027. The AMP review and AMR [57] concluded that seven existing mechanical programmes required updating to confirm compliance with the requirements of RG-0027. Eighteen new programmes were required to support the management of additional ageing and degradation mechanisms. The existing AMP manuals have been updated and new AMP manuals have been developed.

9.5.1.4 Time-Limited Ageing Analyses (TLAAs)

Koeberg design documents and local and international operating experience were utilised to ascertain applicable TLAAs. The methodology was developed and documented in procedure 240-125122792 (*Koeberg Safety Aspects of Long-Term Operation (SALTO) Ageing Management Evaluation Process and Revalidation of the Time-Limited Ageing Analyses*) [39]. The existing TLAAs were revalidated to determine the acceptability of previously analysed structures or components for the planned period of LTO. Existing TLAAs not valid for 60 years of operation are currently being reanalysed to confirm validity for the LTO period. Based on operating experience, new TLAAs were identified, and the analyses for 60 years of operation are currently in progress. These analyses will be completed prior to entry into LTO. Below is a summary of the TLAAs.

- There were 111 TLAAs in total, which included new TLAAs. Of these, 105 TLAAs were revalidated and confirmed to be valid for 60 years. A reanalysis of five TLAAs is in progress. The details of the outstanding TLAAs are contained in Appendix A.1 (*LTO Integrated Implementation Plan*). Although the reanalyses are in progress, it is envisaged that sufficient margins will be available for these components to continue operation for an additional 20 years based on operating experience from EDF. In the event that the reanalyses indicate that adequate margins cannot be maintained for the entire LTO period, ageing management actions required to ensure that the SCCs can meet the design functionality of SSCs are included in Table 9-4.
- The TLAAs were managed according to the design process 331-86 (*Design Changes to Plant Structures or Operating Parameters*) [121]. The analyses were referenced in the updated SAR. The updated SAR contains the current list of applicable TLAAs for the intended period of operation.
- The actions from the TLAAs required to support the conclusions of the analyses were incorporated into the relevant AMPs of the component and managed through the AMPs throughout the plant life, including the intended period of LTO.

- In accordance with RG-0027, if the outcome of a TLAA does not support the 20-year extended life period, then mitigating actions must be taken. This can be the replacement of equipment, or where this is not possible, an ageing management action must be in place. Where the TLAAs have not been completed, these mitigations have been identified. In the case of the EQ and civil TLAAs, the actions have been included in Appendix A. The replacement of the Valcor solenoid valves and RRA 005 and 007 MTs on both units initially formed part of the LTO IPP in 331-618, revision 2. These have been removed from the LTO IPP in this revision since further analyses confirmed that the qualification limitations related to entry into LTO are not applicable and the LTO requirements are met.

Table 9-4*: Actions for TLAAs not Validated for Entire LTO Period*

TLAA Title	Component	Actions
Environmentally assisted fatigue	• Reactor coolant pump • Reactor pressure vessel internals • Main coolant lines • Auxiliary lines • Control rod drive mechanism • Pressuriser heater sleeves	Should the results not support a 60-year life, the following options will be considered: a) Introduction of an augmented inspection scope to address the overstepping components. This is dependent on the location of the component, its accessibility for inspection, and an inspection technique suitable for detecting the expected fatigue initiation in the component. b) A review of transient(s) resulting in the greatest contribution to the overstepping component cumulative usage factor (CUF) to determine whether the transient numbers can be reduced for these specific components c) Replacement of the component, if feasible. This essentially starts the fatigue process anew.
Crack growth analysis of flaws detected in service and fatigue and thermal ageing analysis of manufacturing flaws and flow tolerance	Pressuriser spray nozzles	The study considered the effects of thermal embrittlement, which could reduce the critical crack size and allowable rate of crack propagation for the postulated defect sizes. Should the results not support a 60-year life, the following options will be considered: a) More frequent surface inspections (existing ISI augmented module) b) Defect removal and repairs
Reactor pressure vessel internals (RPVIs) – thermal	Reactor pressure vessel internals	Parts of the RPVI sensitive to flaws include the core barrel upper shell-to-flange weld and core barrel shell welds in the core region. Should the

Table 9-4*: Actions for TLAAs not Validated for Entire LTO Period*

TLAA Title	Component	Actions
ageing and neutron embrittlement		results not support a 60-year life, the following options will be considered: a) Introduction of remote weld inspections (new ISI augmented module) b) Core barrel replacement
Reactor pressure vessel internals – vibration	Reactor pressure vessel internals	The main input to this study was the amount of wear in the upper core plate (UCP) guide pins and the radial keys. This increases support gaps between the RPV and RPVI, with a consequential increase in the loadings (from seismic and LOCA events) to the RPVIs. Should the analysis not support a 60-year life, the action below will be considered: a) Repair or replacement of the worn parts or surfaces

Table 9-5 contains a list of safety analyses (TLAAs) concluded in the SALTO project that have been reanalysed for the additional 20 years of operations and have been submitted in support of the safety case.

Table 9-5: *TLAAs Reanalysed and Processed for NNR Submission*

Item	IGALL Reference	Description of TLAA	Components
1	106	Environmentally assisted fatigue	Pressuriser
2	Additional TLAA	Reactor pressure vessel PWSCC	Reactor pressure vessel
3	Additional TLAA	Reactor pressure vessel	Reactor pressure vessel
4	112	Reactor coolant pump flywheel	Reactor coolant pump flywheel
5	201	Equipment qualification	IC in-core thermocouples cables
			Rotork valve actuators
			Valcor solenoid valves
			EBA AMRI Actuators – Type C AMRI
			Jeumont-Schneider
6	108	Polar crane	Polar crane
7	301	Containment	Strain gauges
			Dynamometers
			Pendulums
			Temperature gauges
			Concrete inspections and repairs

9.5.1.5 Programmatic Aspects of Ageing Management

The section demonstrates that the assertions made below related to ageing management aspects are valid.

- The facility has adequate processes and procedures to ensure that the requirements of sections 6 and 7 of RG0027 are met throughout all plant life-cycle phases, thereby assuring that the effects of ageing are effectively managed throughout the life of the plant, including the LTO period.
- Ageing management has been adequately considered in the design of the facility in accordance with national and international good practices, including ageing management considerations in the plant change management processes.
- The obsolescence management framework is in line with industry standards.

9.5.1.5.1 Ageing Management Programmes and Processes

A self-assessment SE 38545 (*Review of the Interim Regulatory Guide on Ageing Management and Long-Term Operations of Nuclear Power Plant [RG-0027 Rev. 0] Against the Current Plant Ageing Management Processes*) [227] was performed to ascertain gaps in the AM process and procedures for compliance with section 6 of RG-0027. The assessment found a few programmatic-related gaps that required updating of the required process and procedures.

Through the self-assessment, the following ageing management provisions were confirmed or verified to be in place to ensure safe LTO:

- The ageing management standard, 240-149139512 (*Ageing Management Requirements for Koeberg Nuclear Power Station*) [52], provides the overall requirements for the ageing management of safety-related equipment and indicates the links to related physical and non-physical ageing management processes for the current and extended life of the plant. The ageing management standard covers all stages of plant equipment life (that is, design, construction, manufacturing, commissioning, operating, LTO, suspended operation, and decommissioning). The standard also establishes the roles and responsibilities of ageing management activities.
- The AM process 331-275 (*Process for the Development and Control of Ageing Management at Koeberg Operating Unit*) [107] describes the ageing management process, the ageing management review process, the development and review of AMPs, the review of TLAAs, the reporting of AM, and the control and update of the ageing management database throughout the operating life, including the planned LTO period. This document includes roles and responsibilities.

- All the ageing management programmes as defined in the processes mentioned above are required to be consistent with the nine attributes of an effective AMP. If a programme does not meet all of the attributes, its use has been justified and documented.
- The ageing management database contains ageing degradation mechanisms and the ageing effects for in-scope SSCs and how these are managed.
- In line with the requirements of RG-0027, a comprehensive set of ageing management programmes is maintained in document 240-150483693 (*Ageing Management Programmes List*) [53].
- Koeberg implements operating experience processes KAD-025 (*Processing of Operating Experience*) [155], 331-23 (*Processing of Industry Operating Experience in Nuclear Engineering*) [106], and KGA-035 (*Processing of Experience Feedback Received through the EDF Co-operation Agreement*) [171]. These processes are used to periodically evaluate plant and industry-wide operating experience, as well as research and development (R&D) results, and modify any ageing management programmes to ensure the continued effectiveness of ageing management. KAA-688 (*Corrective Action Process*) [146] is used to report any adverse conditions of SCCs important to safety such as failures and non-conformances. The corrective action process is further elaborated in detail in § 9.8.
- A scope-setting process to ascertain SSCs subject to ageing management has been developed and implemented. The process is documented in 240-125839632 (*Koeberg Safety Aspects of Long-Term Operation (SALTO) Scoping Methodology*) [40]. The in-scope AM SSCs have been integrated into the classification database and are controlled by the classification document 331-93 (*Guide for Classification of Plant Components, Structures, Parts, Services and Software*) [123]. The classification software database will automatically classify all new in-scope AM SSCs. Where new non-safety SSCs affect important-for-safety SSCs, the design processes and design checklists will govern the update of the classification database.

The ageing management processes mentioned above have been seen and accepted by the NNR.

The NNR guide RG-0027 describes the programmes discussed below as essential for LTO, and this section describes the programmes and how they are managed at Koeberg. The effectiveness of these programmes was assessed against the nine attributes of RG-0027 in the ageing management self-assessment SE 38545 [227] and the PSR (Equipment Qualification and Ageing) [114], and no significant gaps were found.

- Safety-related surveillance programme

  The Safety-Related Surveillance Manual (SRSM) has been developed and is utilised to manage the surveillance of all safety-related equipment and functions analysed in the SAR. The results of the SRSM tests are compared to previous results to identify degradation that could be related to ageing. The functionality and performance of equipment are monitored periodically to ensure that it remains fit for purpose or to perform its safety function. Equipment upgrades or equivalencies due to obsolescence or ageing are continually initiated where equipment delivers unreliable or out-of-range results. These deficiencies are highlighted by the surveillance programme.

The integrity of the barriers between radioactive material and the environment (that is, the primary pressure boundary and the containment) are addressed in the SRSM under the EPP (airlocks and penetrations containment integrity) and EAS (containment spray) systems containment leak tightness surveillances, which address containment integrity. The RCP (reactor coolant) system and the pressuriser leak tightness and overpressure protection surveillance address protection of the second confinement barrier.

The RPR system (reactor protection and associated protection channels) surveillances are contained in the SRSM, which assures functionality and operability of the reactor protection and safeguard system actuation and operability.

The availability of items whose failure can adversely affect nuclear or radiation safety is ensured by the operating technical specification (OTS) operability requirements, with regard to which the successful SRSM surveillances are instrumental in demonstrating that these requirements are met.

Functional testing to ensure that the tested SSCs are capable of performing their intended function(s) is covered in the SRSM. Where certain functional assumptions are made in the SAR, these assumptions are verified through testing in the SRSM.

The SRSM confirms the provisions for safe operation that were considered in the design and assessed in construction and commissioning and that are verified throughout the operation of the plant.

The SRSM will continue to supply data from monitoring relevant parameters to be used for assessing the service life of SSCs for the planned period of LTO. The periodic monitoring will remain in place to ensure that equipment remains capable of performing its required safety function. In doing so, anomalies related to the ageing of equipment can be picked up when these affect the performance of the equipment. The SRSM verifies safety margins in general.

SRSM assessment against the nine attributes in Annexure A of regulatory guide RG-0027 was included in the scope of the ageing management self-assessment, SE 38545. Further details on how the programme met the requirements in section 6.3.3 of RG-0027 are elaborated in Appendix C.

- Maintenance programme

Koeberg has an effective maintenance programme in place, which is aligned with international standards and guidelines such as Institute of Nuclear Operations (INPO) KAA-913 (*Integrated Equipment Reliability Process*) [154]. The preventive maintenance (PM) programme is dynamic due to it being a living programme with changes based on internal and external plant OE. The

PSR review of the actual condition of SSCs identified the maintenance programme as a strength.

The top-tier document governing the maintenance process is KSM-LIC-001 (*Requirements for the Control of Maintenance*) [188]. The purpose of this document is to define the requirements for the maintenance process and the controls to be in existence to comply with the requirements of the nuclear licence. One of the core aspects of KSM-LIC-001 with regard to the maintenance approach is the need for feedback on equipment conditions, including ageing, for continuous improvement of the programme.

KSA-913 (*Integrated Equipment Reliability Standard*) [182] and KAA-913 (*Integrated* Equipment *Reliability Process*) [154] provide for the preventive maintenance of equipment to high levels of safe and reliable plant operation in an efficient manner and apply to all elements and activities that address the scoping and equipment reliability (ER) classification of components, continued equipment reliability improvement (that is, the preventive maintenance change process), implementation of recurring preventive maintenance tasks, long-term planning and life-cycle management, corrective action for component failures, and performance monitoring.

Feedback on equipment condition is obtained through the corrective action programme failure investigations and monitoring of equipment condition through failure finding tasks as part of the maintenance programme. The frequency of tasks is initially based on original equipment manufacturer (OEM) recommendations but is modified or adapted based on OE, feedback from ageing management programmes, and new inspection methods and techniques.

All failure modes are linked to preventive (and/or predictive) maintenance (PM) tasks in the PM templates. Degradation and ageing effects are included in the PM templates. This is taken into consideration during the development of the component PM strategies in the selection of maintenance activities and frequencies. PM strategies are housed digitally in the preventive maintenance software application. The maintenance execution records are housed on the SAP application.

The assessment of the PM programme was two-fold: firstly, to assess the process and, secondly, to assess the specific PM programme strategies at a component level.

The integrated equipment reliability (ER) process was assessed against the nine attributes in Annexure A of regulatory guide RG-0027. The assessment was performed during the ageing management self-assessment (SE 38545). The purpose was to demonstrate how the requirements in Annexure A, “Generic Attributes of an Effective Ageing Management Programme” of RG-0027, were met. The review identified improvement opportunities and one gap regarding quality management. Actions to resolve the gap and address improvement opportunities were initiated.

The ageing management evaluations compared all SSCs relating to ageing management programmes with the in-scope preventive maintenance strategies on IQReview. This evaluation identified the scope of PM strategies for review.

The identified PM strategy updates are complete.

- SE 35244-049 SE: Koeberg PM templates are to be revised as required to include failure modes and effect analysis to identify the relevant failure modes and failure causes to support the PM tasks identified. The action has been completed for the instrumentation, electrical and mechanical domains.
- CR 116340-016 CA: update the PM programme for the SALTO scope as identified through the mechanical AME review. PM strategies for commodities identified by the mechanical AME review were approved and this corrective action is closed.
- CR 116340-017 CA: validate PM programme strategies on SAP according to the SALTO scope as identified through the mechanical AME review. The validation of the PM strategies was completed and this corrective action is closed.

Furthermore, any PM strategy updates identified during the development of new AMPs will be tracked using equipment reliability change requests (ERCRs). This is in line with the requirements in KAA-913.

Overall, the PSR actual condition of SSC review concluded that the actual condition of SSCs important to safety indicated that all programmes at Koeberg associated with maintaining the condition of SSCs were adequate and well implemented and provided confidence in the delivery of safety functions of SSCs important to safety during LTO, with no significant impact on nuclear safety.

- Water chemistry programme

KSC-003 (*Water Chemistry Programme*) [183] provides for all primary and secondary controls in accordance with approved procedures and chemistry technical specifications. The chemistry programme was assessed using the requirements of the IAEA SSG-13 (*Chemistry Programme for Water Cooled Nuclear Power Plants*) [262].

The assessment of the adequacy of the water chemistry programme did not identify any gaps related to the objective of the programme as it related to ageing management and confirmed that the programme met the regulatory criteria as follows:

- The programme ensured that degradation due to stressors in water chemistry did not have an impact on the ability of SSCs to perform their intended functions in accordance with the assumptions and the intent of the design.
- The water chemistry programme avoided, where possible, and minimised the harmful effects of chemical impurities and corrosion on plant SSCs.

  - The programme was effective in maintaining the water quality required by the technical specifications.
  - The programme specified the scheduling and the analytical methods used to monitor the chemistry and the means of verification of the effectiveness of the chemistry programme.
  - The programme provided the necessary chemical and radiochemical environment to ensure safe LTO and the integrity of structures or components within the scope of ageing management and evaluations for LTO.
  - Additionally, the programme met the nine attributes of an effective AMP as defined by RG-0027 (*Interim Regulatory Guide – Ageing Management and Long-Term Operations of Nuclear Power Plants*) [294].

  Further details on how the programme met the requirements of RG-0027 are provided in Attachment C of KBA0022CHEMJUSTIF2 (*Justification for the Koeberg NPS Chemistry Operating Specifications)* [160].

- In-service inspection programme

  Koeberg has an in-service inspection programme (ISIP) that has been developed and is maintained in accordance with US NRC requirements for in-service inspection as contained in 10CFR50.55a to provide assurance that the structural integrity and operability of SSCs important to safety are within acceptable limits. The ISIP consists of the following documents:

  - 240-110745414 (*Standard for the In-Service Inspection Programme at Koeberg Nuclear Power Station*) [34] provides the requirements for developing the ISIP manual for the fourth and subsequent inspection intervals.
  - 240-119362012 (*Fourth Interval In-Service Inspection Programme Requirements Manual (ISIPRM) for Koeberg Nuclear Power Station*) [36] summarises the fourth interval ISIP requirements for the examination and testing of ASME Class 1, 2, 3, MC, and CC components and component supports as required by ASME Section XI and augmented in-service inspection (ISI) source documents. The ISIPRM was developed in accordance with 240-110745414 (*Standard for In-Service Inspection Programme*) [34].
  - The in-service testing (IST) programme intends to establish assurance on the operability readiness of design safety class components under all design basis conditions. This assurance is established using various periodic surveillances. The operability readiness statement confirms that the design basis safety functions of IST components remain within acceptable limits. Document 240-97087308 (*In-Service Testing Programme Requirements Manual*) (ISTPRM) [85] establishes testing and examination requirements to assess the operational readiness of components important for nuclear safety as listed in the appendices.

As part of the ageing management self-assessment SE 38545 [227] [294], the ISI programme was assessed against the nine attributes in Annexure A of regulatory guide RG-0027 [294]. Further details on how the programme met the requirements in section 6.3.3 of RG-0027 are elaborated in Appendix C.

- Equipment qualification programme

  An equipment qualification (EQ) programme has been implemented at Koeberg to achieve qualification and to maintain the qualified status of in-scope equipment through the life of the equipment. The programme provides assurance that qualified equipment can perform its safety function(s) before, during, and after a design basis event (DBE), such as a loss-of-coolant accident (LOCA), high-energy line break (HELB), main steam-line break (MSLB), design extension conditions (DEC), seismic events, and/or other environments. The effects of significant ageing mechanisms are addressed in the equipment qualification programme. The EQ programme requirements are documented in the following procedures and are based on the NNR guide RG-0027, section 6.3.3 (b). Document 331-186 (*Equipment Qualification Programme Standard*) [100] provides the requirements for establishing and implementing the equipment qualification (EQ) programme at the Koeberg Nuclear Power Station.

  - 331-187 (*Equipment Qualification Process and Responsibilities*) [101] describes the equipment qualification process and responsibilities to maintain the qualification status for all electrical and instrumentation and control (I&C) in-scope equipment, including the period of LTO.

    The EQ programme is consistent with the nine attributes of an effective AMP listed in Annexure A of RG-0027.

    The list of equipment requiring qualification is provided in 240-155832775 (*Equipment Qualification Master List (EQML) for Harsh Environment*) [55]. The list contains all qualified electrical and I&C equipment as defined by SAR II-1.11 (*Environmental Qualification of Electrical Equipment for Accident Conditions in the Containment*) [178] and included in the EQ programme.

    There is no formal EQ programme for mechanical or civil equipment. This is in line with the IAEA standards. However, the equipment qualification requirements and the qualification assessment for mechanical equipment are provided in the engineering nuclear position paper, 240-109728634 (*Environmental Qualification of Mechanical Equipment*) [33]. This position paper provides industry practices relating to equipment qualification of mechanical equipment, the assessment on how Koeberg assures the qualification of mechanical equipment and the justification for not having a formal mechanical equipment qualification (MEQ) programme. The assessment is based on applicable industry OE, the current Koeberg design codes and maintenance practices, and the recommended practices provided in the ASME QME-1 standard (*Qualification of Mechanical Equipment Used in Nuclear Power Plants*) [229].

Requirements are in place to ensure that the qualified life of in-scope equipment is preserved for the full range of specified service conditions. The EQ preservation requirements are provided in the following documents:

- 331-219 (*Environmental Qualification Maintenance Manual for Equipment Located in Harsh Environments*) [105]

  The manual prescribes the requirements to ensure that the qualified equipment and parts are suitably qualified and maintained in their qualification status throughout the plant operating period, including the period of LTO. The environmental qualification maintenance manual (EQMM) lists all qualified equipment and its components or parts that must be maintained or replaced until the end of its qualified life. In accordance with the EQMM, qualified components must be replaced before the end of the qualified life specified in the EQMM, unless a reassessment to extend the qualified life is conducted and justified. The EQMM is reviewed and updated periodically in line with the requirements of the EQ programme on maintenance feedback, plant transients, and operating experience (OE).

- 240-130611911 (*Environmental Qualification Requirements for Safety-Related Equipment Located in Mild Environments*) [41]

  This document contains the maintenance requirements for components located in mild environments. Qualification for equipment located in mild environments is demonstrated by ensuring that equipment meets or exceeds the specified requirements and the specified performance requirements.

  An EQ Manual 331-219 (*Environmental Qualification Maintenance Manual for Equipment Located in Harsh Environments*) [105] and 240-155832775 (*Equipment Qualification Master List (EQML) for Harsh Environment*) [55] have been established to manage and preserve the qualification of qualified equipment located in harsh environments against degradation mechanisms. In accordance with the IEC/IEEE 60780-323:2016 (*Nuclear Facilities – Electrical Equipment Important to Safety – Qualification*) [273], a qualified life is not required for equipment located in a mild environment and which has no significant ageing mechanisms and is operated within the limits established by applicable specifications and standards.

  The equipment qualification time-limited ageing analyses (EQ TLAA), as required by the RG-0027, were performed, and the EQ manual, 331-219 (*Environmental Qualification Maintenance Manual for Equipment Located in Harsh Environments*) [105], was updated with the results of the EQ TLAA. The EQ manual provides the updated qualified life for components with a revalidated life of 60 years and actions that are required to support the qualification of components that were subjected to further reanalysis for LTO and the components that must be replaced prior to LTO. These actions are listed in Appendix A.

The EQ programme manual, 331-219 (*Environmental Qualification Maintenance Manual for Equipment Located in Harsh Environments*) [105] provides actual environmental conditions to be monitored to obtain information necessary for the assessment of ageing effects on the equipment in its actual operating environment. Temperature data and radiation data are available on the InSQL database and from the installed remote measuring devices are trended, and conditions are assessed to determine the impact on the condition of qualified equipment and to identify corrective actions, if necessary. In addition, the requirements of the environmental condition monitoring programme (ECMP), which has been developed for Koeberg, are documented in 240-165386950 (*Environmental Condition Monitoring Programme (ECMP)* [77] *for Electrical Cables and Qualified Equipment*). The ECMP provides requirements for the monitoring of environmental conditions external to the equipment, such as temperature, radiation, and externally induced vibration from mounting points or earthquakes.

The effective review of the equipment qualification programme is carried out periodically and reflected in the programme health report in accordance with 331-148 (*Programme Engineer's Guide*) [97]. The review assesses the nine attributes of the programme, taking into account the effects of ageing on equipment during service and the effects of possible changes in environmental conditions during normal operation.

9.5.1.5.2 Ageing Management Consideration in Design, Design Changes, and Replacements

The plant has been designed in accordance with design codes and standards as discussed in § 9.4.2. These standards have requirements for ageing management embedded in them. Thus, using the standards provides assurance that ageing is considered in the design stage. The design processes provide for any ageing management updates when new SSCs are introduced at the plant during the LTO period.

To maintain the design performance of the plant, Koeberg employs three different plant change processes: the equivalency process 331-143 (*Equivalency Study Process to Change the Plant*) [94], the process 240-86502715 (*Processing Minor Modifications*) [82] for minor modifications, and the design change process 331-86 (*Design Change Process*) [121].

This section discusses how the requirements of RG-0027 related to the consideration of ageing management in design changes are fulfilled.

- The equivalency process

  The equivalency process changes the plant by replacing the existing component with an equivalent or alternative component and maintaining the design basis of the plant. This is done by performing a comprehensive technical evaluation of the functional requirements, requirements for mounting and interfacing to the plant, and functional requirements for normal, transient, and accident conditions. Apart from the technical study, the equivalency study also considers previous operating experience (local and international), new failure modes and effects analysis, obsolescence, and ageing degradation effects.

Documents 331-143 (*The Equivalency Procedure to Change Plant*) [94] and 331-144 (*Standard for the Preparation of an Equivalency Study*) [95] require the determination of any potential impact of ageing or degradation effects on new proposed equivalent components.

The standard addresses the requirement of assessing ageing effects and degradation mechanisms for new proposed equivalent components important to safety. Document 331-144 provides all the requirements when replacing an equivalent or alternative component on the plant. The requirement is incorporated into the equivalency procedure 331-143, indicating the responsibilities for reviewing and finding ageing degradation mechanisms for new equivalent components and updating the ageing management documents with the new information. Document 331-155 (*The Equivalency Guide*) [98] explains how the ageing management requirements are addressed during the application of the equivalency process. Document 331-412 (*Equivalency Check Sheet*) [110] shows ageing management as one of the considerations during the equivalency process.

- The design process and minor modification process

The design process changes either the physical plant or the theoretical plant to maintain or improve the performance and standards of the plant. The objective of the minor modification process is to make non-complex plant changes, which the Maintenance Department can execute under a maintenance budget. The design change process is used when complex and large changes are made to the plant that requires a multidisciplinary team led by a project manager to manage the commercial aspects of the project, the project development and execution, and any project elements that affect nuclear safety. Although the criteria and process of the two plant change processes differ, the technical design content to consider is the same. The detailed design document is divided into four parts. Part A details the technical aspects of the new design, its calculations, and the impact on the plant environment, operation, and processes. Part B details the installation and interfacing to the plant. Part C details the selection of the new SSCs to be installed. Part D is the configuration management section, which indicates all documents, processes, and programmes that must be updated because of the design change. Therefore, ageing management is mainly considered in Parts C and D of the detailed design.

The form 331-211 (*Design Input Consideration Checklist*) [104], design standard 331-83 (*Requirements for Plant Changes Affecting the Design of Koeberg*) [119], 331-86 (*Design Procedure*) [121], and design guide 331-87 (*Design Engineering Guide*) [122] include the determination and mitigation of ageing effects and degradation on all new SSCs introduced at the plant.

Document 240-86502715 (*Processing Minor Modifications*) [82] includes determining and mitigating ageing effects and degradation on all new SSCs introduced at the plant.

The design standard 331-83 (*Requirements for Plant Changes Affecting the Design of Koeberg*) [119] and guide 331-87 (*Design Engineering Guide*) [122] address the consideration of design margins and design features to facilitate ageing management effects when performing modifications to the plant.

In section 3.10 of document 240-143890978 (*Design Template*) [48], the appropriate maintenance basis and ageing management programmes are considered for all new designs.

During the design stage, ageing effects are considered for design basis conditions, transient conditions, and postulated initiating conditions by selecting appropriate equipment for the design. In 240-143890978, guidelines are provided for correct equipment selection. Document 331-496 (*Equipment Qualification File Template*) [111] is completed during the design stage, considering the environmental operating and accident conditions and all qualification test reports for the equipment qualification programme. Procurement specifications are compiled using specification standard 331-165 (*Nuclear Specification Standard*) [99] to ensure that correct specifications and materials of equipment are obtained to support the applicable ageing management programmes.

TLAAs are ascertained using the criteria documented in 240-132364298 (*Initial List of Time-Limited Ageing Analyses for Koeberg Nuclear Power Station*) [43]. The requirements when performing analyses or reanalyses for TLAAs are documented in design guide 331-87 (*Design Engineering Guide*) [122] and design procedure 331-86 (*Design Changes to Plant, Plant Structures or Operating Parameters*) [121].

Fundamentally, all the processes mentioned above require determining design specifications, including specifications related to the management of ageing effects, such as equipment qualification specifications.

In conclusion, based on the design processes mentioned above that are in place, the Koeberg plant design process is deemed to be adequate for LTO.

9.5.1.5.3 Technological Obsolescence Management Programme (TOMP)

With modern technology, the problem of obsolescence has grown rapidly and has become a major concern for all nuclear power stations.

Koeberg has implemented TOMP 331-146 (*Process for the Technological Obsolescence Management Programme*) [96] to manage the technological obsolescence of all SSCs important to safety, including the associated spare parts.

The TOMP was developed in accordance with the requirements of RG-0027 and establishes the key structures that support the management of the obsolescence programme, including the roles and responsibilities for determining obsolescence, prioritisation of the obsolescence, and solution development. The PSR ageing review identified a deviation with a "low" significance grading for the TOMP. The deviation was raised relating to the proactive implementation of the technological obsolescence programme. The safety improvement for this deviation is appropriately ranked and included in the LTO IPP (Refer to § 14.0.), which has since been implemented to resolve this deviation.

9.5.2 Ageing Management Risk Items

The LTO assessment identified that some SSCs important to safety had degradation mechanisms that are presently not adequately managed. If not rectified, these may affect safe LTO. The section discusses the ageing risks associated with these SSCs and the mitigations to ensure safe LTO.

9.5.2.1 Containment Building

In the early 2000s, the buildings were subject to chloride-induced reinforcement corrosion. The corrosiveness of the local environment was not sufficiently addressed in the original design of the structures. The risk of chloride-induced reinforcement corrosion was, therefore, not fully mitigated.

Mitigation efforts since the discovery include a modification to the containment buildings which is included in the LTO IP. The proposed modification is to implement an impressed current cathodic protection (ICCP) system into the concrete of the containment buildings to neutralise the corrosion effects of the chlorides. The modification aims to delay the corrosion process, extend the lifespan of the structure, and maintain the integrity of the third barrier for the period of LTO. A mock-up of the ICCP is planned to provide additional confidence in the application of the ICCP on the containment buildings.

Despite the ageing risk mentioned above, the containment buildings are at present acceptable for operation based on current surveillance monitoring results. They will be able to perform their design safety function as the third barrier. The monitoring is continuous and effective at detecting ageing effects. Completed refurbishments have been effective. Refurbishments are scheduled when ageing effects are detected. Safety analysis (time-limited ageing analysis) found that the structural integrity of the containment buildings was ensured for the planned long-term operation period.

The design of the containment structures at Koeberg is similar to the French 900 MW PWR series. The concrete structures are monitored and are required to meet the criteria of ASME XI, subsection IWL [2.1.5], with some exceptions due to the cement grouting of the tendons, which makes it impossible to comply with the prescribed method of inspection as described in ASME XI. Especially, the inspections foreseen for greased tendons according to ASME XI and NRC 1.35 [2.1.6] are not possible in the case of cement-grouted tendons. Hence, the monitoring programme includes ASME XI requirements and French practices. General acceptance criteria for pre-stressed containment monitoring are stated in the OECD guide [2.1.3], Chapter 8.2.3.10, Table 0.1, consistent with the

EDF routine maintenance programme in RCC-G [2.1.4]. The acceptance criteria for pre-stressed containment monitoring are stated in SAR II-1.9.2.5.2.1 (*Post Tensioned Concrete*) [178].

A 10-yearly integrated leak rate test (ILRT) and continuous operational licence-binding activities are conducted on both containment buildings. The ILRT is an all-encompassing test conducted on the containment buildings to determine the leak tightness of the buildings. The ILRT also tests the structural integrity of the containment buildings. The ILRT increases the pressure inside the containment buildings up to 400 kPa (gauge), representing the pressure the containment buildings would experience during a loss-of-coolant accident. During the pressurisation and depressurisation of ILRT, the containment buildings are closely monitored to determine their behaviour and condition. SAR II-4.2.2.2 (*Test Description, Acceptance Criteria*) [178] documents the ILRT leak tightness and structural integrity acceptance criteria. The leak tightness acceptance criteria are based on the US NRC 10CFR50 Appendix J. The containment test criteria are documented in the 'SRSM' (KBA0022SRSM00000) [164] under the EPP system.

During the last ILRTs in 2015, both the Unit 1 and 2 containment buildings demonstrated the expected behaviour and were qualified as suitable for operation. Figure 9-5 shows the containment global leak rate test results for both units. The ILRTs provided confidence that the degradation of the reinforced concrete had not compromised the structural integrity of the containment buildings. The monitoring of the containment buildings during the ILRTs also showed that the refurbished areas behaved uniformly with the remainder of the concrete. The results for the containment structure are documented in correspondence DB2015-0020 (*System Design Engineering Acceptance of the Unit 1 ILRT [Outage 121] Structural Integrity Results*) [128] for Unit 1 and correspondence DB2016-0002 (*System Design Engineering Acceptance of the Unit 2 ILRT [Outage 221] Structural Integrity Results*) [129] for Unit 2. Complete detailed results for the containment structure and leak rates of the ILRTs are stored at Koeberg's documentation centre and can be provided on request. Essentially, an ILRT on a containment building removes any uncertainty surrounding the integrity of the structure. Therefore, the test will be performed to confirm the expected behaviour of the buildings in accordance with the ISI program requirements. Additionally, a detailed assessment of the containment structures has been documented in 331-623 (*Engineering Position on Containment Structures for Long-Term Operation*) [112] which further elaborates that no structural concerns were identified. Refer to K-28880-E. The next test is scheduled for x27 and is included in the LTO Implementation Plan in Appendix A.2.

As part of the ageing management evaluation in preparation for LTO, a technical evaluation of the containment buildings was conducted. The evaluation included a revalidation of time-limited ageing analysis (TLAA) assumptions. The TLAA for the containment buildings was "revalidated" for 60 years. The IAEA safety report and guides relating to TLAA documents were considered in conjunction with this process. The TLAA concluded that containment structural integrity was ensured for the planned long-term operation from a concrete compression and structural integrity perspective. However, the TLAA included recommendations surrounding the functionality of the civil monitoring system and inspections and repair measures. These recommendations were not considered limiting and did not prohibit the operability of the containment buildings for the additional 20 years. The recommendations made in the analysis were incorporated into the AMP.

Considering the continuous assessments that the plant conducts (that is, inspections and TLAA), mitigations that have been completed (that is, repairs), and improvements embarked on (that is, modifications), the current condition of the buildings is deemed to have sufficient integrity, and the design of the buildings remains fit for purpose and suitable for long-term operation.

Figure 9-5: *Containment Global Leak Rate Test Results*

9.5.2.2 Nuclear Island: Aseismic Bearings

The aseismic bearings are situated between the upper and lower raft of the nuclear island at 12 m below MSL, known as the aseismic vault. It has been designed to act as a filter between the soil and the SSCs of the nuclear island in the event of an earthquake. The material composition used for the aseismic bearings is low-damping neoprene rubber, known as a low-damping rubber bearing (LDRB) type. A monitoring programme of the bearings has been implemented since the commissioning of the plant and is defined in the SAR II-1.9.1.8 (*Monitoring Programme*) [178]. The monitoring programme, monitoring results, and safety improvements to ensure that the aseismic bearings are acceptable for LTO are discussed below.

1. **Tests conducted on sample bearings and in-situ bearings**

   In conformance with the requirements of the nuclear licence (NIL-01 Variation 19), a series of tests and visual inspections are required to be performed on the aseismic bearings (and sample bearings) to demonstrate that specific parameters remain within the design basis. These are performed on representative samples in a laboratory and the actual bearings at prescribed intervals. These tests include:

- visual inspection (in-situ bearings);
- shore hardness measurements of neoprene (in-situ bearings);
- distortion measurements and the regreasing of selected bearings (in-situ bearings);
- a friction test (sample bearings); and
- shear modulus tests (sample bearings).

The inspections and tests are performed as documented in formal procedures and using authorised inspectors and qualified laboratories.

2. Test results of the inspections and tests

Visual inspections: the number of defects seen on the wrapping had been increasing due to ageing. Defects on the steel plates were superficial, and defects related to the grease seal did not show any corrosion of the interface of the friction surfaces. To date, no significant operability-related defect has been found on bearings.

Shore hardness**:** the average values obtained indicated small changes from the originally installed values (68 Shore A in 1978 versus 70 Shore A in 2016). SAR II-1.9.1.5.3.6.3 (*Neoprene Elastomer)* [178] specifies the shore hardness for neoprene on the bearings as being between 63 and 70 Shore A.

Distortion measurements: between 1 mm and 4 mm distortions were recorded, which had not significantly changed since 1978. The maximum permissible distortion was 50 mm as per the technical manual. The measurement is a crude method and is intended to identify gross irregularities.

Friction: the values obtained during the coefficient of friction tests (circa 2000) were higher than the design specification. As a result, a seismic reanalysis of the nuclear island was performed using increased friction values. The results indicated a minimal change in the response of the structures if no sliding was permitted (SAR II-1.9.1.8 (*Monitoring Programme*) [178]). Therefore, requirements related to coefficient of friction tests were subsequently removed from the monitoring programme.

Shear modulus tests: the average static (limit 0,95 ± 15%) and dynamic (limit 1,3 ± 15%) shear modulus of the sample test bearings, as observed from the last three test series, appeared to be dropping over time. Two static shear modulus test results and four dynamic shear modulus test results were marginally outside their lower limits. No results were outside their upper limits. A lower static or dynamic shear modulus was not expected to negatively affect the seismic response of the nuclear island.

Delamination of the neoprene from the reinforcing steel plates was detected. This concern was analysed by finite element methods and a limit to the extent of the delamination was determined. It was not an ageing phenomenon, but rather a manufacturing-induced defect [178].

EDF has experienced an increased shear modulus. The only other plant with seismic bearings is Cruas, whose bearings consist only of neoprene components (no sliding plates). The EDF results confirmed the ultimately expected plateau of around a 37% increase in shear modulus. This value still respects the original design assumptions. Although the Koeberg shear modulus results are not of concern, given the EDF results, further evaluation of available literature, test data and operating experience has been completed to understand the ageing characteristics of the bearings. Important industry guidelines for ageing management were also benchmarked against Koeberg's ageing management programme. The outcome of the evaluation was favourable and confirmed that the aseismic bearings are fit for purpose. The results are documented in 331-645 (*Elastomeric aseismic bearings – Current position and the way forward*) [117] and 331-675 (*Overview of the ageing management programme for the aseismic bearings*) [118].

3. **SALTO ageing management review outcomes**

As part of the SALTO ageing management assessment project, Eskom utilised a consortium of the original architects of the plant (or original equipment manufacturer (OEM)) and a local engineering firm to benchmark all safety-related equipment against the internationally accepted IAEA International Generic Lessons Learned (IGALL) to identify enhancements that could be made to the management of ageing. For the bearings, there was little benefit due to the uniqueness of the equipment. Nevertheless, given the OE discussed above, improvements in the ageing management programme have been identified and incorporated

4. **Aseismic bearing safety improvement**

Koeberg has monitored the seismic bearings according to the relevant regulations, and all results and findings have been submitted to the NNR. There has been only limited maintenance (plate interface seal replacement) required. The visual and test results of the in-situ and sample bearings have shown no significant change in material properties. Based on these facts, the bearings can perform their design functions and remain fit for purpose and suitable for long-term operation. This has been confirmed following an evaluation of available literature, test data and operational experience to better characterise the material properties of the bearings. The results of the evaluation and benchmark of industry guidelines for aseismic bearing AMPs have informed the Koeberg aseismic bearings AMP. The aseismic bearings will continue to be tested in accordance with its AMP (incorporating the NNR approved in-service inspection programme) and supplemented, as necessary, with additional testing during LTO.

9.5.2.3 Ageing Management of Switchboards and Cables

The main function of the plant switchboards and cables is to maintain the continuity of electrical power supply to the relevant loads during normal and abnormal unit operation and for safe shutdown during normal and accident conditions (SAR II-10 (*On-site Electrical System*) [178]).

LTO requires an evaluation of the plant switchboards, switchboard components, plant cabling condition, and suitability for safe operation up to 60 years of commercial operation, with some of the plant equipment requiring operating beyond 60 years for purposes of plant decommissioning.

An initial study in 2009 found that large-scale plant cabling and switchboard replacements and refurbishments could be required for LTO. This was largely based on the obsolescence of switchboard spares and the unknown condition of the plant cables. The subsequent re-evaluation of the plant switchboard and cable condition, as well as suitability for LTO, is documented in 32-T-IPDK-008 (*Koeberg Switchboard, Switchboard Components and Plant Cabling Evaluation for LTO*) [92], which includes the findings of the recent SALTO ageing management assessment and PSR. The contents of this document form the basis of the arguments presented in this section relating to the ageing of the cables and switchgear for safe LTO.

Nuclear industry operating experience and experience from similar plants at EDF indicate that large-scale switchboard and cable replacements are normally not required for LTO.

Large-scale switchboard and cable replacements for Koeberg's LTO are therefore not anticipated because of the robust design, equipment operating conditions, spare availability and low failure rates, as discussed below.

The equipment scope considers all the Unit 1, Unit 2, and Unit 9 6,6 kV (LGi and LHi), 380 V (LKi and LLi), 220 V (LNE and LMA), and DC (LAi, LBi, LCi, and LDA) switchboards, along with the respective MV (6,6 kV), LV (380 V, 220 V, and DC), and I&C cabling, both environmentally qualified (EQ) and non-EQ-related (SAR II-10.3.1 (*6,6 kV Switchboards*); II-10.3.2 (*380 V Switchboards*); II-11.2.2 (*I&C Power Sources*); and II-11.3 (*General Installation – Cabling and Connections*) [178]).

Most switchboards are located in the electrical building. A few switchboards are located in areas such as the nuclear auxiliary building and the emergency diesel generator rooms. The electrical cabling is distributed throughout the plant, with most cabling located on perforated or enclosed cable trays.

For 6,6 kV power cables, the same specifications apply to safeguard and non-safeguard cables containing cross-linked polyethylene (XLPE) insulation, with the environmentally qualified (EQ) cables requiring unique specifications and either ethylene-propylene rubber (EPR) or XLPE insulated cores. Similarly, for the LV and I&C cables containing copper cores, the insulation is polyvinyl chloride (PVC), with the EQ cables containing EPR insulation (SAR II-11.3.4 (*Electrical Cables*) [178]).

The evaluation 32-T-IPDK-008 (*Koeberg Switchboard, Switchboard Components, and Plant Cabling Evaluation for LTO*) [92] did not specifically evaluate all the individual codes and standards applicable to the switchboards, switchgear, switchboard components, and cables. However, a high-level analysis was performed of the applicable codes and standards documented in the plant safety review (PSR plant design review) and the SAR.

The evaluation of the latest revision of these codes and standards did not find any requirements related to plant switchboards, switchboard components, or plant cabling that would affect nuclear safety and LTO.

Availability of major spares such as circuit breaker and contactor modules for both the 6,6 kV and 380 V switchboards has been confirmed, with Koeberg already having purchased sufficient 6,6 kV circuit breakers for LTO.

The re-evaluation of the initial study of 2009 determined that large-scale plant cabling and switchboard replacements or refurbishments were not anticipated, based on the following:

9.5.2.3.1 Design Adequacy of Switchboards and Cables

The Delle-Alsthom switchboards have a robust design, with no indication of any significant deterioration in the condition of the switchboards. The availability of switchboard component spares ensures that switchboard components that no longer meet their design tolerances are replaced or refurbished in good time according to normal routine maintenance practices.

The assessment of the plant cables (MV, LV, and I&C) confirmed the conservative design of the cable installation and operating conditions. Due to the installation of cables with higher voltage ratings, the electrical stress on the cable insulation is low. Increased cable core sizes overcome the problem of cable heating and voltage drops on cables with long cable runs. This design margin significantly lowers the risk of cable ageing concerns.

Large-scale cable and switchboard replacement for LTO due to ageing is not a standard practice in the nuclear industry and is mainly done on a case-by-case basis, depending on the cable or switchboard condition or specific requirements, such as equipment obsolescence or fire resistance requirements of cables.

The French electricity utility EDF has plants and electrical components similar to Koeberg in design and materials. In some cases, the EDF plants have the same switchboard types and plant cabling as Koeberg. EDF is not planning any large-scale cable replacements for its plants that exceed 40 years of operation. The EDF Saint-Alban nuclear power plant, commissioned in 1985 to 1986, has the same Delle-Alsthom-manufactured switchboards and switchgear as Koeberg and is one of the EDF plants scheduled for operation beyond 40 years; it is not planning switchboard replacements.

There have been Koeberg switchboard and switchboard component failures, either during normal plant operation or during routine maintenance and surveillance. However, the number and type of failures do not indicate significant switchboard condition deterioration. Issues with obsolete equipment, such as the degrading protection relays, are being addressed through 331-146 (*Technical Obsolescence Programme*) [96].

Koeberg has experienced a few cable failures mainly due to maintenance practices or cable installation practices playing a significant role in the failure mechanism.

All equipment failures are investigated through the corrective action programme (CAP), and appropriate corrective actions are implemented, including updating the maintenance basis, if required, to prevent a reoccurrence.

The EDF ageing management matrix was used to develop 240-101650256 (*Koeberg Ageing Management Matrix*) [30]. A review of 240-156945472 (*SALTO Ageing Management Assessment Report (Interim)*) [57] confirmed that all in-scope plant electrical switchboards, switchboard components, and plant cabling were comprehensively addressed and aligned with the IAEA IGALL requirements. Where the AMP requirements for switchboards, switchboard components, and plant cabling were found to require additional alignment or improvement, the appropriate ageing management programmes or manuals were updated, or new ones were compiled. The full Koeberg cable ageing management programme (CAMP) and the associated environmental condition monitoring programme are in the process of being implemented.

The Koeberg CAMP includes the standard 331-127 (*Cable Ageing Management Programme at Koeberg Operating Unit*) [93], the procedure 331-198 (*Cable Ageing Management Programme Roles and Responsibilities at Koeberg Operating Unit*) [103], and the cable ageing management manuals prepared for the MV, LV, and I&C cables, based on the requirements stipulated in RG-0027 [294]. The scope of the CAMP is provided in 240-166828385 (*Cable Master List*) [78] and includes the in-scope cables.

During the revalidation of the EQ TLAA, the qualified life of EQ cables was limited to the end of the current licensed term (40 years). The subsequent reanalysis considered additional qualification information and concluded that the cables are qualified for the full period of LTO.

9.5.2.3.2 Switchboards

The Koeberg life of plant plans (LOPPs) for the 6,6 kV, 380 V, 220 V, and DC switchboards provide comprehensive information on the status of the plant switchboards. The LOPPs confirm the respective switchboard condition, equipment reliability, and spares availability. Additionally, equipment monitoring, testing, known or potential threats, spare availability, and obsolescence are also documented in the LOPP. Therefore, based on the operating experience contained in the LOPP, it is concluded that the service life of the switchboards will reach the required 60 years without adverse impact on nuclear safety.

Routine maintenance, monitoring, and testing of switchboards and components may uncover equipment out of tolerance (for example, circuit breaker timing tests) or degraded (for example, control relays). The frequency of switchboard surveillance and maintenance or testing varies depending on the component. The current switchboard maintenance tasks are aligned with industry good practices recommended by EPRI, the US NRC, INPO, IAEA, and EDF. New inspection and testing techniques have been introduced to detect possible failure mechanisms, such as partial discharge testing for switchboard insulation degradation; refer to KBA-0022-N-NEPO-LOPP-068 (*Plant Engineering Life of Plant Plan 6,6 kV System Maintenance Regime*) [162].

Document 240-164966115 (*Ageing Management Programme for Switchboards, Associated Switchgear Components and Metal*) [76] evaluated switchboard ageing management. It analysed the potential age-related degradation mechanisms for the electrical switchgear (circuit breakers and contactors), metal enclosures, and switchboard components such as control cabling, relays, insulators, and protection relays. The evaluation confirmed the adequacy of the switchboard monitoring, testing, and maintenance regime. According to the IAEA ageing management programmes (AMPs), additional requirements for both maintenance procedures and maintenance basis programme changes have been added to align and improve the current maintenance programme.

Switchboard replacements are currently not expected for LTO due to observed equipment reliability and availability of spares. Should any switchboard component obsolescence or reliability issues that cannot be addressed through the obsolescence or modification process arise later during plant operation, then limited switchboard replacements could be pursued to release spares to address these concerns. Obsolete items such as the currently obsolete 6,6 kV and 380 V switchboard protection relays and fuses will continue to be replaced as required using the technical obsolescence programme.

The evaluation performed during the SALTO project confirmed the good condition and reliability of the switchboards, the availability of switchboard spares, and that a comprehensive ageing management programme was in place.

Ongoing periodic testing and operations of the equipment provide confidence in the operability and reliability of the switchboards.

Switchboards and their components are expected to continue performing their design functions and remain available and reliable for continued safe operation.

9.5.2.3.3 Cables

The ageing of power cables is mostly a result of adverse localised environments and adverse service conditions, with the most common contributors being heat and wet environments. With water-induced cable degradation causing accelerated ageing of power cables found in environments susceptible to moisture or water ingress, Koeberg initiated a comprehensive cable ageing management programme (CAMP) in 2013, initially focusing on the 6,6 kV power cables, which were the most susceptible to ageing. The low-frequency (0,1 Hz) tan δ cable testing method was adopted at Koeberg as the most appropriate cable test method for all cables. Based on the CAMP programme, to date, testing results have not found any indication of 6,6 kV cable degradation that would preclude LTO.

Testing of the LV and I&C cables has been limited, with a focus on visual and tactile inspections. 240-165386950 (*Environmental Condition Monitoring Programme (ECMP) for Electrical Cables and Qualified Equipment*) [77] monitors harsh environments and revises the testing and inspections as required.

EQ cables are included in the CAMP programme. Additional EQ requirements are addressed in 331-186 (*Equipment Qualification Programme*) [100].

The operability of the switchboards and associated electrical cabling is also assured by being permanently energised. In addition, frequent swapping of electrical trains and periodic testing confirm the operability of the switchgear, switchgear components, and associated cabling.

Of concern is the long-term operation of wetted XLPE-insulated power cables. These cables can develop water trees, and as these water trees progress, they can result in cable failure. A comprehensive CAMP is in place to manage the ageing of the Koeberg cables, with an environmental condition monitoring programme being prepared to support the CAMP. The initial focus of the CAMP is wetted MV power cables. Included in this focus are wetted cable samples to be sent for further laboratory ageing testing.

Large-scale cable replacements are not anticipated for LTO due to the current cable reliability and condition and industry operating experience.

9.5.3 Ageing Management of SSCs Supporting Licence-Binding Programmes

A review of the Koeberg licensing basis manual (KLBM) revealed that certain SCCs not included in the scope of SSCs important to safety are used in support of certain licence-binding programmes. While these SSCs did not play a direct role in managing fundamental plant safety functions, they were included in regulatory programmes. Consequently, it was deemed necessary to review the activities and governance processes that ensured their continued reliability during LTO. A comprehensive assessment was conducted and documented in 331-602 (*Assessment of Out-of-Scope Structures, Systems, or Components used in other Regulatory Programmes*) [113].

The above mentioned SSCs are used in the following licence-binding programmes:

- Radiation protection (RP)
- Emergency planning (EP)
- Environmental monitoring (EM)
- Chemistry monitoring
- Licensed operator training (full-scope operating simulator)
- Nuclear security (the nuclear security report was provided in a separate and confidential submission)

The ageing management assessment of the SSCs concluded that:

- processes related to the ageing management of these SSCs are controlled and approved in line with the established quality management system;

- the station executes the maintenance, surveillance, inspections, testing, and quality control activities of the equipment on a predetermined schedule;
- defects are reported for resolution after surveillance, maintenance, inspections, or testing. Unexpected events and failures are raised, evaluated, and tracked within the station's corrective action process;
- equipment performance is monitored, and availability is trended to ensure planned equipment replacements. In the event of an unplanned equipment failure, redundant and diverse equipment is available on-site;
- most equipment are off-the-shelf items; however, if the equipment lacked spare parts or has no technical support from suppliers or industry, the equipment is added to the technological obsolescence management programme (TOMP). (Refer to 331-146 (*Technological Obsolescence Management Programme*) [96].) After that, it was managed within this process;
- the Quality Assurance Department generates a cyclical audit schedule annually in line with the quality assurance monitoring schedule for the power station, which includes assessments of the governance of the programmes, including the equipment used in the execution of these programmes. This provided assurance regarding adherence to that governance; and
- each licence-binding programme has its own set of requirements, documents, and reliability objectives.

Therefore, the ageing management of the SSCs used in support of the licence-binding programmes is governed by procedures that ensure the long-term reliability and availability of the equipment.

9.5.3.1 Radiation Protection Programme

The Koeberg documents for the governance of, and guidance on, this equipment in the radiation protection programme include the following:

- 238-54 (*Radiation Protection Licensing Requirements for Koeberg Nuclear Power Station*) [26]
- 238-36 (*Operational Radiation Protection Requirements*) [12]
- 238-19 (*Generation Division Radiation Protection Manual*) [7]
- 238-42 (*Radiation Dosimetry Requirements*) [16]
- 238-44 (*Requirements for Radiological Surveillance Instrumentation*) [18]
- 238-49 (*Liquid and Gaseous Effluent Management Requirements*) [22]
- KAA-679 (*Control of the Measuring and Test Equipment at Koeberg Nuclear Power Station*) [145]
- KAA-584 (*Radiation Instrument Management*) [136]

The typical equipment used in this programme includes, inter alia, the ARGOS 6, the GEM 5, and the whole-body counter.

9.5.3.2 Emergency Planning Programme

The Koeberg documents for the governance of, and guidance on, the equipment used in the emergency planning programme include the following:

- KAA-811 (*The Integrated Koeberg Nuclear Emergency Plan*) [148]
- KAA-611 (*Emergency Mustering, Accountability and Evacuation*) [139]
- KAG-001 (*Emergency Exercise Management and Assessment*) [156]
- KEP-I-002 (*Operating Instructions for Emergency Plan Communications Equipment*) [165]
- KEP-I-014 (*Operation, Use and Availability of Emergency Plan Portable Instrumentation and Equipment*) [166]
- KAG-003 (*Maintenance and Inventory Control of the Emergency Management Facilities and Equipment*) [157]

The typical equipment included in this programme is the portable diesel generators, jellyfish catch nets, oil boom, emergency vehicles, and bubble curtains.

9.5.3.3 Environmental Monitoring Programme

The Koeberg documents for the governance of, and guidance on, the equipment used in the environmental monitoring programme include the following:

- 238-47 (*Radiological Environmental Survey Requirements*) [20]
- KAA-597 (*Environmental Surveillance Programme*) [138]
- KAG-006 (*Koeberg Meteorological Programme*) [158]
- 238-52 (*Emergency Planning Meteorological Requirements for Nuclear Installations*) [24]
- 235-54 (*Radiological Protection Licensing Requirements for Koeberg Nuclear Power Station*) [4]

The typical equipment included in this programme is the meteorological equipment mounted on the 50 m and 120 m masts and handheld air measurement devices.

9.5.3.4 Chemistry Monitoring Programme

The Koeberg documents for the governance of, and guidance on, the equipment used in the chemistry monitoring programme include the following:

- KAA-595 (*Control of Chemistry Instrumentation, Analysers, and Equipment*) [137]
- KAA-640 (*Control of Items Leaving Site for Repair or Service*) [144]
- KFC-AC-008 (*Instrument Defect Form*) [167]
- KFQ-ML-001 (*Equipment Control Form*) [168]
- KWC-AC-002 (*Maintenance and Service of Analytical Chemistry Laboratory Equipment and Instrumentation*) [189]

The typical equipment included in this programme is the chemistry portable and laboratory equipment, Crison CM 35, Orbisphere 3650 dissolved oxygen analyser, Metrohm 912 conductometer, and Nova TRI gas analyser.

9.5.3.5 Licensed Operator Training (Full Scope Operating Simulator)

The Koeberg documents for the governance of, and guidance on, the equipment used in the licensed operator training programme include the following:

- KAA-503 (*Modifications to Simulator*) [135]
- KAA-857 (*Management and Oversight of the Full Scope Operator Training Simulators at Koeberg Nuclear Power Station*) [152]
- KFT-072 (*Simulator Availability*) [169]
- KGT-025 (*Simulator Maintenance, Access, Operation and Initial Conditions and the Training and Authorisation of Simulator Operators*) [174]

The typical equipment included in this programme is the simulator's electronic display screens.

9.6 Radiation Protection

Eskom is committed to ensuring that nuclear and radiation safety receives the highest priority to provide for the protection of persons and the environment against harmful ionising radiation in accordance with the safety principles and requirements addressed in Document 32-227 (*Radiation Protection and the Safety of Radiation Sources Policy*) [88].

Koeberg complies with the dose limits and dose constraints stipulated in the regulations on safety standards and regulatory practices, R.388 [241]), the requirements on risk assessment and compliance with principal safety criteria (RD-0024 [289]), and the radiation dose limitation at Koeberg, RD-0022 [288]).

This section discusses the adequacy of the current radiation protection (RP) programme to ensure compliance with the regulatory requirements mentioned above and to support LTO. The scope of the radiation protection programme was assessed during the PSR safety performance review, and it was concluded that there are adequate processes, procedures, guides, and work instructions to support LTO. Therefore, this section demonstrates that:

- the established RP programme is adequate (that is, mature, effective, and efficient) for the current licensing basis and for the period of LTO;
- the dose to workers and the public will be kept as low as reasonably achievable (ALARA);
- there is an effective RP programme to support the effluent management programme;
- the RP organisation is sufficiently resourced to implement the RP programme;
- the RP training is effective and adequate to ensure the necessary level of competence for personnel having functions relevant to the protection and safety of members of the public;
- the RP programme is adequate to support the safe off-site transport of radioactive material, and the programme remains adequate for the period of LTO;
- the process for managing radiation generators and radioactive sources is adequate; and
- there are adequate processes and procedures to conduct formal investigations of abnormal conditions (relating to radiation protection) that can arise associated with Koeberg operations, including applying operating experience (OE).

This section does not discuss the impact of LTO on radioactive waste management and the environment, as this is discussed in § 9.7.3 and § 9.7.4 of this document.

9.6.1 Radiation Protection Programme

A radiation protection programme has been developed and implemented, and the programme is effective in ensuring that the exposure of workers, the public, and the environment to radiation is minimised. The programme has established processes and procedures for the radiological protection of workers, the public, and the environment, both on off-site. The programme incorporates local and international requirements. The aspects considered in the programme include:

- RP organisation;
- access control;
- radiation area classification;
- radioactive source control;
- radiation exposure:
    - the ALARA programme; and
    - the dosimetry programme;
- radiation worker training;

- the radioactive waste management programme; and
- the radioactive effluent management programme.

The SAR [178] provides an overview of the protection of workers against radiation during normal operation in terms of 'General Safety Principles Addressing the Radiological Consequences of Normal Operation' (SAR I-4.2), 'Biological Shielding' (SAR III-5.1), 'Airborne Contamination' (SAR III-5.2), 'Radiation Monitoring' (SAR III-5.3), the 'Radiation Protection Programme' (SAR III-5.4), and 'Radiological Design Criteria and Features' (SAR III-5.5). Requirements for protecting workers during planned exposure situations are documented in the RP standard [26].

The effectiveness of the RP programme was assessed during the PSR safety performance review [67], and it was concluded that the programme is well documented and implemented. The PSR review also found that the associated processes and procedures for protecting workers, the public, and the environment, as well as the roles and responsibilities for protection and safety, are documented and implemented [67].

The RP programme is regularly reviewed for alignment with the latest industry developments in the radiation protection field and, therefore, remains valid to support LTO. Independent inspections and audits of the RP programme are performed by the Nuclear Safety Assurance and Quality Assurance Departments and during peer reviews (WANO) to ensure that the programme is properly implemented in accordance with regulatory and international requirements.

9.6.1.1 Radiation Protection Organisation

The RP organisational arrangements were reviewed during the PSR safety performance review and were found adequate to support LTO. Functional responsibilities and educational and training (credential) requirements for RP positions are listed in KSH-010 (*Functional Responsibilities for Radiation Protection at Koeberg Operating Unit*) [185]. Although the RP organisation has experienced high rates of attrition in recent years, it has not affected the mandate of the RP organisation (which is mainly the implementation of the RP programme), and the organisation has embarked on a recruitment campaign to replenish the resources. Newly appointed resources are trained and authorised in accordance with RP training programme requirements, as discussed in § 9.6.1.2.

9.6.1.2 Radiation Worker Training

The RP standard, 238-43 (*Requirements for Radiation Workers*) [17], requires that a training programme for radiation workers be implemented. Workers are trained in accordance with RPG-001 (*Radiation Worker Training Course*) [226] and registered as radiation workers prior to granting them access to a radiological controlled zone.

The training programme for RP personnel and radiation workers ensures that only authorised personnel can perform and supervise activities that result in radiation exposure. This ensures adherence to ALARA practices to prevent unplanned exposures. Radiation worker training provides, among others, the following:

- Requirements for radiation workers and special persons at Koeberg
- Aspects of radiation, specifically ionising radiation
- Radiation units and biological effects, including the risk of radiation
- Radiation detection and protection
- Contamination detection and protection
- Control of materials inside the radiological controlled zone
- Radiation worker practices

On successful completion of the training, employees are issued with dosemeters and other protective equipment. Radiation worker training and authorisation are tracked through RadPro software, which is an integrated software used for tracking the training status and verifying the authorisation of radiation workers for entry into radiation control zones. Procedures KAA-634 (*Responsibilities for the Radioactive Material Control Programme*) [142], KSH-010 (*Functional Responsibilities for Radiation Protection at Koeberg Operating Unit*) [185], and KSH-012 (*Radiation Protection Standards and Expectations*) [187] provide the basis for radiation worker training. The NNR requirements for the education, training, and responsibilities of personnel performing radiation protection-related functions are documented in KSH-010 [185]. Procedure 238-34 (*Optimisation of Radiation Protection*) [10] requires that Eskom radiation workers receive initial and follow-up training. Radiation protection training is managed according to KGT-055 (*General Radiation Protection Training Guide*) [175] and KGT-056 (*Radiation Protection Department Training Programme Guide*) [176].

9.6.1.3 Radiation Area Classification

Radiological controlled zones (areas) are classified according to the requirements in 238-54 [26], 238-36 [12], and KAA-637 (*Access Control to Radiological Controlled Zones*) [143]. Administrative and physical controls employed to manage red zones and RP-locked zones are described in KWH-S-043 (*Control of Red Radiation Zones and Radiation Protection Locked Zones*) [202]. Radiation zones are classified based on the highest dose rate from any component or structure in the area. There are four radiation zone classifications:

- **Green zone**: the general area radiation dose rate is less than 25 µSv/h.
- **Yellow zone**: the general area radiation dose is greater than, or equal to, 25 µSv/h by less than 1 mSv/h.
- **Orange zone**: the general area radiation dose rate is greater than, or equal to, 1 mSv/h, but less than 10 mSv/h.

- **Red zone**: the general radiation dose rate is greater than, or equal to, 10 mSv/h.

The administrative controls for containment entries at the red zone or the radiation protection locked zone conditions, including emergency entries into all controlled zones, are documented in KWH-S-025 (*Containment Entries at Red Zone or Radiation Protection Locked Zone Conditions Including Emergency Entries into all Controlled Zones*) [198]. Airborne contamination zones are classified in terms of the concentration of airborne radionuclides in the area in accordance with KWH-S-015 (*Airborne Contamination Surveys*) [196]. The boundaries of controlled zones are demarcated and signposted in accordance with KWH-S-048 (*Signposting and Barricading in Radiological Controlled Zones*) [206]. The procedures used for the management and control of sources are KAA-633 (*Control of Radioactive Sources and X-Ray Equipment*) [141], KWH-S-007 (*Leakage Tests on Sealed Radioactive Sources*) [195], and KWH-S-041 (*Radiation Protection Source Control*) [201].

9.6.1.4 Entry into Radiation Control Zones

Entry into radiological areas is controlled by means of the use of radiation worker certification in accordance with procedures KAA-637 (*Access Control to Radiological Controlled Zones*) [143] and KWH-S-021 (*Access Control*) [197] and administered by the access control portion of RadPro software. The PSR safety performance review concluded that the process for controlling entry into radioactive areas remained adequate to support LTO.

9.6.1.5 Radiation Source Control

Eskom has developed, implemented and maintains processes, procedures, guides, and work instructions to ensure compliance with document 32-227 (*Radiation Protection and the Safety of Radiation Sources Policy*) [88]. Radioactive sources at Koeberg are categorised in line with IAEA-TECDOC-1344 (*IAEA Standard for Categorization of Radioactive Sources*) [271] and are documented in procedure 238-46 (*Requirements for the Safety, Security and Control of Radioactive Sources*) [19]. The movement of a source between storage areas or from a storage area to a work area is tracked using logbooks. Physical security measures for sources are implemented by security personnel. The storage areas for radioactive sources are approved by the radiation protection officer (RPO – source control) commensurate with the hazard. Measures implemented for ensuring the control of radioactive sources were assessed in the PSR safety performance review. The review concluded that the measures are adequate for LTO, and no deviations were raised [67].

The following documents are used for the management and control of radioactive sources:

- 238-38 (*Radiation Protection Requirements for Baggage Inspection X-Ray Devices*) [13]
- 238-39 (*Requirements for the Safe Use of Industrial Gauges Containing Radioactive Sources*) [14]
- 238-40 (*Radiation Protection Requirements for Industrial Radiography*) [15]

- 238-46 (*Requirements for the Safety, Security and Control of Radioactive Sources*) [19]
- KAA-633 (*Control of Radioactive Sources and X-Ray Equipment*) [141]
- KWH-S-007 (*Leakage Tests on Sealed Radioactive Sources*) [195]
- KWH-S-041 (*Radiation Protection Source Control*) [201]
- KWH-S-045 (*Radiation Protection Requirements for Industrial Radiography on Site*) [203]
- KWH-S-046 (*Radiation Protection Requirements for Use of Soil Moisture and Density Gauge Sources*) [204]

9.6.2 Radiation Exposure

Exposure to radioactive material can be a result of normal operations or accident conditions. Koeberg has processes and procedures designed not only to comply with the regulatory dose requirements prescribed in RD-0022 (*Radiation Dose Limitation at Koeberg Nuclear Power Station*) [288], but also to minimise radiation exposure of workers, the public, and the environment to levels that are ALARA.

In order to comply with RD-0022, for normal operations, radiation exposure dose limits are documented in 238-35 (*Radiation Protection Dose and Risk Limits*) [11], and for emergency exposure, the limits are documented in KAA-811 (*Integrated Koeberg Nuclear Emergency Plan*) [148].

The dose constraints and dose limits referred to above are as follows:

1. For public dose:
   - Public dose constraint: 0,25 mSv/a.
   - Public dose limit: 1 mSv/a.
2. For occupational exposure:
   - An average effective dose of 20 mSv/a averaged over five consecutive years (within the five-year period, a maximum effective dose of 50 mSv for any single year).
   - An equivalent dose to the lens of the eye of 150 mSv/a; an equivalent dose to the extremities (hands and feet) of 500 mSv/a.

To achieve the objective of minimising radiation exposure, Koeberg has developed and implemented the ALARA programme. The programme is discussed further in section § 9.6.2.2. The PSR safety performance review verified that processes and procedures used to manage and control radiation doses to workers are in compliance with the dose limits specified above and concluded that these procedures and processes are adequately implemented [67].

9.6.2.1 Occupational Exposure

For occupational exposure, relevant dose constraints are used in the optimisation of protection and safety standard 238-34 (*Optimisation of Radiation Protection*) [10]. The RadPro software is used to track occupational exposures. The RadPro software manages data relating to workers in radiological controlled areas; it collects data recorded by electronic dosemeters and checks radiological controlled area access.

Koeberg maintains a radiation dose register for every occupationally exposed worker, and dose records are archived and made available to the NNR. Visitors are informed of applicable radiation protection requirements in accordance with KAH-002 (*Radiation Surveillance Programme*) [159] and KAA-637 [143] prior to entry into radiological controlled areas or a supervised radiological area.

Requirements for records of radiation exposure, dosimetry, functional responsibilities at Koeberg, and the surveillance programme are met and compliance with these requirements is managed in accordance with the documents listed below.

- 32-227 (*Radiation Protection and the Safety of Radiation Sources Policy*) [88]
- 32-226 (*Radiation Protection and the Safety of Radiation Sources*) [87]
- 238-19 (*Generation Division Radiation Protection Manual*) [7]
- 238-42 (*Radiation Dosimetry Requirements*) [16]
- KSH-008 (*Radiation Protection Records, Data, and Information Management*) [184]
- KSH-010 (*Functional Responsibilities for Radiation Protection at Koeberg Operating Unit*) [185]
- KAH-002 (*Radiation Surveillance Programme*) [159]

The maximum occupational dose for the period between 2010 and 2020 was below 20 mSv per annum, and the average occupational dose from 2010 to 2020 was less than 1 mSv per annum (below the 4 mSv per annum dose target). Low collective doses are attributed to the low fuel failure rate, good control of the primary and auxiliary system chemistry, design considerations, a good radiation protection programme, and mainly the effectiveness of the ALARA programme.

Dose assessment is implemented through the implementation of the following procedures:

- KSH-011 (*Radiation Protection Certificate (RPC) Programme Requirements*) [186]
- KWH-B-014 (*Dosimetry Quality Control Programme*) [191]
- KWH-B-015 (*External Dosimetry Control*) [192]
- KWH-B-016 (*Operation, Calibration and Use of The Koeberg Whole Body Counters and the H-3 in Urine Analysis Programme*) [193]
- KWH-B-017 (*TLD Processing and Dose Record Updating*) [194]

- KAA-632 (*ALARA Programme*) [140]
- KWH-AL-004 (*Radiation Protection Formal ALARA Programme Criteria, Actions and Documentation*) [190]
- Utilising the RadPro software

Requirements for unplanned exposures are detailed in KWH-S-015 (*Airborne Contamination Surveys*) [196]. Unplanned exposures are reported to the NNR in accordance with KLA-005 (*Koeberg Event Classification and Reporting Criteria Listing*) [177]. In addition, requirements to manage medical emergencies, radiation casualties, frequency of medical examinations, and psychological evaluations are documented in KSA-055 (*Requirements for the Medical and Psychological Surveillance and Control Programme*) [180].

9.6.2.2 ALARA Programme

An ALARA programme has been implemented to ensure that activities giving rise to radiation exposure are controlled to minimise the radiation dose. The programme is documented in KAA-632 (*ALARA Programme*) [140] and KWH-AL-004 (*Radiation Protection Formal ALARA Programme Criteria, Actions and Documentation*) [190].

The ALARA concept is applied to activities ranging from day-to-day operations to major design changes and plant modifications. The optimisation of protection begins at the planning stage and continues through the stages of scheduling, preparation, implementation, and feedback. Oversight of the ALARA programme and monitoring of its implementation is done with the appointment of ALARA dose champions and the allocation of accountability to management to monitor and report back on effective doses. KWH-AL-004 (*Radiation Protection Formal ALARA Programme Criteria, Actions and Documentation*) [190] contains an extensive list of measures that should be considered for protection and safety during the ALARA review of a task and ALARA items for design/modification reviews. The implementation of the programme is supported by monitoring exposures per individual, group, or task and in comparison, with an extensive set of targets. In addition, responsibility for the implementation and accountability for doses at various levels of the organisation are given. A station ALARA committee provides oversight of the programme and monitors exposures against targets. This ensures that effective doses are kept as low as reasonably achievable.

The ALARA programme was assessed during PSR and was found to be comprehensive and adequate for LTO.

9.6.2.3 Dosimetry Programme

Koeberg has established and implemented a dosimetry programme that is consistent with regulatory and international requirements for monitoring radiation exposures of radiation workers. The dosimetry programme was assessed during the PSR safety performance review and was found to be adequate. Occupational radiation exposure is monitored using electronic personal dosemeters (EPDs) and thermoluminescent dosemeters (TLDs).

In the case of noble gas exposure, the dose is assessed by converting activity concentration to dose using conversion factors and exposure times. Internal dose is assessed using a whole-body counter (WBC). A bioassay programme is used to assess the dose from tritium- or alpha-emitting radionuclides. Records collected as a result of the programme are uploaded to the national dose register and reported quarterly and annually to the NNR. Radiation dose records are available for the workers on a local area network.

To adequately record and track worker exposures, the following procedures are used:

- KSH-008 (*Radiation Protection Records, Data, and Information Management*) [184]
- KWH-B-014 (*Dosimetry Quality Control Programme*) [191]
- KWH-B-015 (*External Dosimetry Control*) [192]

Requirements for the dosimetry programme are contained in the following documents:

- 32-227 (*Radiation Protection and the Safety of Radiation Sources Policy*) [88]
- 32-226 (*Radiation Protection and the Safety of Radiation Sources*) [87]
- 238-19 (*Generation Division Radiation Protection Manual*) [7]
- 238-42 (*Radiation Dosimetry Requirements*) [16]
- 238-35 (*Radiation Protection Dose and Risk Limits*) [11]
- 238-48 (*Thermoluminescence Dosimetry Requirements*) [21]
- 238-54 (*Radiation Protection Licensing Requirements for Koeberg Nuclear Power Station*) [26]

Responsibilities and accountability for the dosimetry programme processes are in accordance with KWH-B-015 (*External Dosimetry Control*) [192], and the programme is implemented in accordance with KWH-B-014 (*Dosimetry Quality Control Programme*) [191], KWH-B-016 (*Operation, Calibration and Use of The Koeberg Whole Body Counters and the H-3 in Urine Analysis Programme*) [193], and KWH-B-017 (*TLD Processing and Dose Record Updating*) [194].

9.6.2.4 Radiation Dose to the Public and the Environment (Normal Operations)

Koeberg has processes and procedures to ensure that effluents released to the environment during normal and anticipated operational occurrences do not exceed regulatory limits. Exposure of the public and the environment to radiation due to ionising radiation from effluent released into the environment is maintained as low as reasonably achievable. Implementation of regulatory requirements associated with public exposure is contained in 238-35 (*Radiation Protection Dose and Risk Limits*) [11].

A retrospective assessment of the dose to the public and the environment is performed and reported to the NNR quarterly and annually. The environmental monitoring programme routinely assesses the dose based on actual nuclide discharges measured at the station. The dose assessment to the public includes exposure pathways (ingestions, inhalation, direct exposure), details around radionuclides that were released (that is, liquid and gaseous effluents), and the member of the critical group/representative person. The public doses resulting from effluent discharges between 2015 and 2020 were below the dose constraint of 250 µSv/a and were less than 1% of the dose limit of 1 mSv/a prescribed by the SSRP, Regulation R.388.

Due to LTO and in accordance with the NNR approved methodology, a prospective dose assessment was performed to determine the potential radiological impact on the public. The prospective dose was assessed using a conservative source term, which considered the potential cumulative radiological impact on the public, including proposed facilities (for example, TISF) that were planned to be built on site. The assessment was performed utilising software (PC-CREAM software). The verification and validation of the software was conducted, and a report was submitted to the NNR, refer to letter K-29456-E.

PC-CREAM software is the primary tool used to model the environmental transfer and calculate the dose for the representative person utilising the consequences of releases to the environment assessment methodology (CREAM).

The prospective dose calculation was performed based on the methodology of NSIP04129 (*A Revised Methodology to Assess the Ionising Radiation Dose for Members of the Public from Normal Operation at the Duynefontyn Site*) [217]. This included:

- radionuclide selection and source terms based on the updated activity migration model source term, taking into account normal operations, including anticipated operational occurrences (AOOs);
- atmospheric dispersion and meteorological data preparation for the PC-CREAM PLUME model;
- an evaluation of soil characteristics in the Duynefontyn regional environment and selection of the appropriate input for use in PC-CREAM Granis and terrestrial transfer factors;
- dispersion of liquid discharges to the sea and bioconcentration and sediment distribution factors for use in PC-CREAM DORIS;
- consideration of short-term or batch discharges;
- build-up of radionuclides in the environment;
- exposure pathways; and
- habit data and defining a representative person.

The results obtained from the prospective source term considering environmental build-up historically and for the next 20 years showed that the dose to a member of the critical group/representative person is estimated to be approximately 94 µSv/a [216], which was below the dose constraint of 250 µSv/a as stipulated in R.388 [241] and RD-0022 [288]. The retrospective results also demonstrated that the dose to the public based on actual discharges is significantly below the dose constraint of 250 µSv/a. The outcomes of the assessment showed that LTO would not result in undue risk to the public.

In addition to assessing the dose to the public, the impact of radiological effluents on the environment was assessed using the ecological risk from ionising contaminant assessment (ERICA) software. ERICA software was developed to assist the user in formulating the problem (involving stakeholders if appropriate), performing an impact assessment, evaluating data, keeping records, and performing the necessary calculations to estimate dose rates for selected biota. The ERICA software has a built-in list of reference organisms. Each reference organism has its specified geometry and is representative of either the terrestrial, the freshwater, or the marine ecosystem. The ERICA software is described in Chapter 7 of the DSSR [216].

ERICA software is used widely to assess the ecological risk associated with the release of radionuclides into the environment in order to ensure appropriate environmental management decision-making. Predicted dose rates can be compared with dose rates known to cause biological effects in non-human species.

The results of the ecological risk assessment for marine and terrestrial reference organisms concluded that the liquid and gaseous discharges from the facility are unlikely to pose a significant risk to the environment [216]. Although the NNR has not issued a regulatory limit on non-human biota, the results showed that the dose rate for both terrestrial and marine environments is well below the dose rate guideline values as applied internationally (that is, 10 µGy/h for marine organisms and 40 µGy/h for terrestrial organisms).

9.6.3 Transport of Radioactive Material

Radioactive material is transported off-site in accordance with the requirements of the IAEA regulations for the transport of radioactive material SSR-6 (*Regulations for the Safe Transport of Radioactive Material Specific Safety Requirements*) [268]. Based on the requirements of SSR-6, the responsibilities and accountability for the transport are documented in KAA-634 (*Responsibilities for the Radioactive Material Control Programme*) [142], KWH-S-033 (*Processing and Administration of Solid Radwaste*) [199], and KWH-S-037 (*Classification of Solid Radioactive Materials and the Acceptable On- and Off-Site Packaging Requirements for Such Materials*) [200]. Below are procedures for the on-site transportation of radioactive material that aim to minimise the staff's radiological exposure.

- KAA-634 (*Responsibilities for the Radioactive Material Control Programme*) [142]
- KWH-S-033 (*Processing and Administration of Solid Radwaste*) [199]

- KWH-S-037 (*Classification of Solid Radioactive Materials and the Acceptable On- and Off-Site Packaging Requirements for Such Materials*) [200]
- KWH-S-047 (*Implementation of the Radioactive Material Control Programme*) [205]

In accordance with the conclusion of the PSR safety performance review, these procedures remain adequate to support LTO.

9.6.4 Operating Experience

The radiation protection programme is informed by internal and external operating experience (OE). Reporting of radiation protection events is in accordance with KAA-688 (*Corrective Action Process (CAP)*) [146] and KLA-005 (*Koeberg Event Classification and Reporting Criteria Listing*) [177]. The following RP procedures are used to manage events:

- KSH-012 (*Radiation Protection Standards and Expectations*) [187]
- KGH-004 (*Radiation Protection Management of Operating Experience Feedback*) [172]
- KGH-010 (*Radiation Protection Response to Incidents/Alarms*) [173]

In accordance with the conclusion of the PSR safety performance review, these procedures remain adequate to support LTO and allow for continuous improvement of the radiation protection programme and practices.

9.7 Impact of Long-Term Operation on Other Operational Safety-Related Programmes

Regulatory requirements for LTO include assessing the impact of LTO on other safety-related programmes as defined in RG-0027 (that is, safety-related programmes other than the ageing safety-related programmes). This section provides information on the impact of LTO on these programmes. The extended operation of the plant has minimal impact on these other safety-related programmes mainly because the current design and operation of the plant predominantly remain unchanged during the period of LTO.

9.7.1 Impact of Long-Term Operation on Nuclear Security

The nuclear security assessment was conducted. The nuclear security review report and its results, including the impact of LTO on nuclear security and the justification for continued operation related to nuclear security submitted to the NNR as a stand-alone confidential submission, due to the sensitivity of the assessed information.

9.7.2 Impact of Long-Term Operation on Emergency Planning

The section discusses the adequacy of emergency planning and response at Koeberg and the impact of LTO on the emergency plan. To assess the impact of LTO on the emergency plan, the emergency plan technical basis was reviewed using the requirements of RD-0014 (*Emergency Preparedness and Response Requirements for Nuclear Installations*) [287] and concluded that the current emergency planning zones (EPZs) remain adequate for the period of LTO. Therefore, LTO would have no impact on the emergency plan. Adequate measures to protect the public in the event of a radiological emergency have been implemented. The facility has effective emergency preparedness and response capabilities at a national and international level.

Koeberg has developed, implemented, and maintains processes, procedures, guides, and work instructions to ensure compliance with the requirements for developing and implementing emergency preparedness and response arrangements. The *NNR Act 47 of 1999* [242] requires that provisions for emergency planning be in place to effectively respond in the unlikely event of nuclear accidents or radiological emergencies. The emergency preparedness and response plan is documented in administrative procedure KAA-811 (*Integrated Koeberg Nuclear Emergency Plan* (*IKNEP)*) [148], which complies with the NNR requirements for emergency preparedness and response (*RD-0014*).

The PSR emergency planning review assessed the adequacy of emergency planning and response arrangements and concluded that these are adequate, appropriately documented, and executed to ensure the protection of workers, the public, and the environment and support LTO [72]. The review also concluded that deviations raised do not pose any risk to LTO. The deviations raised primarily relate to the following:

- The emergency planning zones and arrangements might be inadequate when assessed them against RG-0020 and GSR Part 7, as the technical basis for emergency planning did not consider multi-unit accident conditions. (This was raised in both the PSR's PSA and EP reviews.)
- There were inadequate arrangements and emergency plan staffing resources to maintain the functionality of the emergency response for the failure of both units simultaneously or following severe on- and off-site infrastructure damage scenarios.
- The pager system was unreliable in notifying emergency response organisation staff.

All the deviations mentioned above were graded to have a “low” or “drop” safety significance. Safety improvements to address these deviations are contained in the PSR IIP. (Refer to § 14.0.)

This section demonstrates the following related to the emergency plan:

- The size of the EPZ is adequate for the emergency plan.
- Adequate measures have been put in place to ensure effective emergency preparedness and response to nuclear and radiological emergencies, and such measures will remain in place and adequate during the LTO period.

- In case of a nuclear emergency, appropriate agreements that can be implemented in a coordinated and timely manner with local, regional, national, and international stakeholders are in place. These agreements will remain in place to support LTO.

9.7.2.1 Emergency Plan and Procedures

An effective emergency preparedness and response plan and procedures that consider any action or source that can cause nuclear damage or that can give rise to an emergency requiring intervention have been developed for the facility. The requirements for an emergency plan are contained in standard 238-53 (*Emergency Preparedness and Response Requirements for Nuclear Installations*) [25]. The IKNEP [148] was assessed in the PSR emergency planning review. The review concluded that the content of the IKNEP is in line with regulatory requirements and international standards, and no deviations were raised related to the adequacy of the IKNEP [148].

The IKNEP is informed by a comprehensive safety analysis undertaken to ascertain all exposure sources and evaluate radiation doses associated with the facility as contained in the EPTB. This analysis includes accident scenarios on which emergency planning zones and associated arrangements for emergency preparedness and response are established.

The adequacy of the current emergency preparedness and response plans was assessed in the PSR emergency plan review. The scope of the review included inter alia, the assessment of the content and effectiveness of exercises, the emergency training, the competence of the emergency response organisation, the required functional capability of equipment (including communications equipment), and the adequacy of emergency planning.

The outcome of the review confirmed the adequacy of emergency planning and response arrangements. The review confirmed that the IKNEP was adequate and appropriately documented to support emergency response. The assessment established that:

- emergency plans, policies, and procedures are established, periodically reviewed, and maintained in a state of preparedness and are subject to rigorous routine reviews and technical audits;
- emergency plans are generally well established and reviewed according to predetermined schedules;
- the facilities and equipment used in the emergency planning are deemed adequate to perform their functional requirements;
- most of the requirements for interaction protocols are adequately implemented by Koeberg and the City of Cape Town (CoCT);
- no significant anomalies are evident in the competency of the relevant support organisations with regard to their emergency management functions; and

- overall, the requirements to solicit support organisations, clearly assign their responsibilities, ensure their competence, and have procedures and processes to ensure their support and co-operation effectively and efficiently are met when viewed against international, regulatory, and internal targets and criteria.

The PSR review identified four deviations graded as “low”. The proposed safety improvements are contained in the PSR IIP as discussed in § 14.0.

Current land use, demography, and adjacent sea use were assessed in sections 5.4, 5.5, and 5.6 of the updated DSSR. The updated DSSR also assessed the projected changes in land use, demography, and adjacent sea use for the LTO period (that is, from 2024 until 2045). The assessment included the current and projected nearby transportation and industrial and military facilities around Koeberg. This information is used to assess factors around the site that can impede the implementation of the emergency plan. Thus far, the DSSR update studies have not found any factors around the site that can impede the implementation of the emergency plan.

Updated evacuation time estimates are important for off-site protective action strategies. An updated analysis is required of the time to evacuate various sectors and distances within the EPZ plume exposure pathway for transient and permanent populations, using the most recent demographic and census data. The CoCT is updating the current evacuation time estimates (ETEs) based on the latest population and traffic data. The current evacuation time estimates remain valid until the updated traffic evacuation model report has been finalised, reviewed, and accepted by the NNR.

The EPTB was reassessed based on the requirements of RD-0014 [287]. The conclusion of the EPTB was that the current PAZ remains adequate for implementing precautionary actions to avoid or minimise severe deterministic health effects on the public. The current UPZ distance of 16 km remains adequate for implementing protective actions such as evacuation, sheltering, and iodine prophylaxis to reduce the risk of stochastic health effects to the public for internal events (including internal fire and internal flood events) [224].

Regarding the overall EPZ, the current long-term protective zone distance of 80 km remained adequate to conduct monitoring to ascertain areas where early protective actions such as evacuation or relocation might be necessary to reduce the risk of stochastic health effects from the long-term exposure to deposition of radioactive material for internal events (including internal fire and internal flood events). The sensitivity analysis, which simplistically increased the initial core inventory used as input in PC Cosyma by 10%, showed and concluded that the potential impact of the thermal power uprate (TPU) and steam generator replacement (SGR) projects on the current size of the PAZ (5 km) and UPZ (16 km) is expected to be negligible; that is, no change is expected [224].

Although the EPTB was reviewed based on RD-0014 and found to be adequate, it was also reviewed against RG-0020 and GSR Part 7 requirements (specifically the multi-unit events) and was updated to include multi-unit events, PSA23-0004-R00 (*Proposed EPTB for Multi-Unit Event at KNPS*), which also addresses the PSR’s EPTB related deviations. The updated EPTB was submitted to the NNR for approval.

The IKNEP incorporates organisations such as:

- the City of Cape Town;
- regional, provincial, and national disaster management teams;
- local supporting organisations such as Necsa, the NNR, the South African Police Service, etc.; and
- international support from the IAEA, OEM (Areva), and EDF.

9.7.2.2 Emergency Exercises and Training

Emergency exercises are performed on a regular basis by both Eskom and the NNR to evaluate the effectiveness of all aspects of the response plan such as:

- training and authorisation of the emergency response organisations (annual reauthorisation);
- evaluation of the adequacy and functionality of the equipment;
- implementation of the protective actions; and
- effectiveness of the coordination among the various stakeholders.

The outcomes of exercises are recorded and gaps in excellence are identified to determine areas for improvement, thus ensuring continuous learning. A five-year exercise plan is maintained to ensure that all aspects of the plan are thoroughly practised. The training of emergency response personnel was assessed during the PSR emergency planning and found to be adequate. However, the PSR found that there was insufficient evidence that the damage controllers and operations support centre supervisors underwent training or periodic retraining on the use of non-permanent equipment used when responding to accidents more severe than design basis accidents. The deviation was graded “low”, and safety improvement actions are included in the PSR IIP to address this deviation.

The effectiveness of the emergency exercise was assessed in the PSR emergency planning review. The assessment concluded that it is adequate and effective. The effectiveness will continue to be assessed on a regular basis during the LTO period, and where gaps are identified, normal Eskom processes will be followed in line with regulatory requirements, as well as international standards, to ensure that measures are put in place to resolve the areas for improvement.

9.7.2.3 Facilities and Equipment

Koeberg has emergency response facilities that serve as a central location for coordinating response team activities. The adequacy of the facilities and equipment was assessed in the PSR emergency planning review. The review of the equipment and facilities included physical walk-downs of relevant on- and off-site areas. The review of facilities and equipment used in the emergency planning deemed them adequate to perform their intended functional capabilities to support nuclear or radiological emergencies [148].

9.7.2.4 Identification, Notification, and Activation

The IKNEP provides for the identification, classification, public notification, and activation of emergency response teams, both on and off site. The emergency classification system used at Koeberg is aligned with the NNR requirement stipulated in RD-0014 [287]. The emergency classes used are:

- unusual event;
- alert;
- site emergency; and
- general emergency.

The requirements for identifying operational anomalies, proper classification of such events, and notification and activation of the appropriate elements of the emergency response organisation(s) were assessed during the PSR emergency planning review. Included in the review was the assessment of the provisions for transport accidents involving radioactive waste or fresh nuclear fuel that took place outside of the site.

The PSR review concluded that Koeberg has developed and implemented processes and procedures that are aligned with both local and international requirements. These processes and procedures would remain valid for the period of LTO. Should any improvements be required, Eskom will follow appropriate established processes to implement changes in compliance with NNR requirements.

9.7.2.5 Off-Site Arrangements

In accordance with the *NNR Act 47 of 1999* [242], Eskom has entered into an arrangement in the form of a memorandum of agreement (MoA) with the Western Cape Government and the City of Cape Town. In addition, Eskom has entered into other arrangements in the form of contracts, partnership, and affiliations to support the IKNEP.

The off-site arrangements were assessed during the PSR emergency planning review. The review concluded that the overall requirements associated with the off-site arrangements are in line with regulatory and international standards and practices and supported implementation of the IKNEP. The off-site arrangement would remain in place for the intended life of the plant.

9.7.2.6 Monitoring and Assessment

Koeberg has developed strategies for monitoring radiological conditions at the plant and in the environment during an emergency. The strategies allow for appropriate decisions to ensure the safety of the public during a severe nuclear accident. The strategy for monitoring and assessment of conditions at the plant and in the environment during an emergency was reviewed during the PSR emergency planning review, and it was concluded that the strategies are adequate for LTO. Any changes to strategies, methods, or standards during the LTO will be made in line with approved Eskom processes and regulatory requirements.

9.7.2.7 Accident Mitigation and Protective Actions

The EPTB was reviewed, and it was found that Koeberg has adequate strategies for protective actions and accident mitigation (evacuation, sheltering, food banning, iodine, and relocation). Intervening organisations have been identified and authorised to respond appropriately to a nuclear emergency. Emergency planning arrangements included procedures and communication protocols for contacting relevant response organisations such as firefighting, medical, and police. At each stage of an emergency, a graded approach was applied commensurate with the hazard assessment and the protection strategy.

The accident mitigation and protective actions were assessed during the PSR emergency planning review and found to be adequate for LTO.

9.7.2.8 Developments around the Site

Koeberg has adequate processes and systems to monitor developments around the site that may affect emergency planning arrangements. Development around the Koeberg site is managed in line with Cape Town's Municipal Spatial Development Framework (MSDF), which sets out the spatial vision and development priorities for Cape Town. This is informed by requirements of the Spatial Planning and Land Use Management Act 16 of 2013 (SPLUMA). Eskom has established the Koeberg Licensing and Liaison Forum (KLLF), which serves as an interested and affected party that monitors, reviews, and screens the potential impact of proposed developments around the site.

The Emergency Planning, Steering, and Oversight Committee (EPSOC) is a regular interface meeting between several stakeholders, including Eskom, the NNR, CoCT (various departments), Western Cape Government (various departments), South African Police Service, South African Broadcasting Corporation, Department of Mineral Resources and Energy, and Department of Public Enterprises. The scope of the meeting includes the evaluation of the effects of any recent residential and industrial developments around the site.

During the LTO period, Koeberg will continue to ensure that proposed developments around the site do not pose a significant impediment to the current emergency preparedness and response plan through the implementation of measures as listed above. The organisation will also continue to assess opportunities that can be used to further enhance existing measures to manage developments around the site.

9.7.2.9 Emergency Response Organisation

The emergency response organisation responsible for manning the emergency control centre (ECC) and the technical support centre (TSC) was assessed in the PSR emergency planning review and was found to be adequate to support the emergency plan. The emergency response organisation has the means, resources, and tools necessary to support protective actions in the emergency planning zones. However, the PSR also found that, for external events affecting both units, staffing might be inadequate to maintain the functionality of the emergency response, and that there are inadequate arrangements to maintain adequate functionality of the TSC and ECC post severe onsite and offsite infrastructure damage. Considering the low risk of an event affecting both units, these deviations were graded "drop" and "low", respectively. The safety improvements to address these deviations are included in the PSR IIP.

The roles and responsibilities of all resources, including various organisations involved in implementing the emergency response, are documented in KAA-811 (*Integrated Koeberg Emergency Plan*) [148]. LTO will not affect emergency planning and response. Koeberg has adequate resources (both internally and externally) required to implement and maintain the emergency plan.

9.7.3 Impact of Long-Term Operation on Radioactive Waste Management

Koeberg generates gaseous, liquid, and solid radioactive waste with varying levels of radioactivity, including high-level radioactive waste, such as spent nuclear fuel. Radioactive waste needs to be safely managed to protect human health and the environment in the present and the future [239]. Section 7.3 (2) of the regulatory guide RG-0027 [294] requires that a decision to pursue LTO be based on an evaluation that addresses radioactive waste management for LTO.

The section discusses the current arrangements for radioactive waste management, the effectiveness of such arrangements, and the impact of LTO on the arrangements. This section only addresses the impact of LTO on solid radioactive waste; the impact on liquid and gaseous radioactive waste is discussed in § 9.7.4. The radiation protection aspects relating to the handling, storing, and transporting of radioactive waste are covered under § 9.6.

The implemented processes and programmes for the safe management of radioactive waste comply with national and international standards for radioactive waste management. Provision has been made for additional waste storage capacity for the period of LTO, and the operating regimes of the facility will remain unchanged.

9.7.3.1 Radioactive Waste Programme at Koeberg

The effectiveness of the radioactive waste management programmes was reviewed during the PSR safety performance review [67], and compliance with regulatory requirements was verified. This review assessed whether Koeberg had implemented programmes for the minimisation and safe management of radioactive waste. Although additional radioactive waste will be generated during the LTO, the average annual radioactive waste volume produced is not expected to increase as the operating processes will remain largely unchanged. However, the cumulative volume of radioactive waste produced will increase due to the extended period of operation, i.e., LTO. Koeberg will continue to monitor the best available techniques/methodologies to improve the programme to ensure compliance with regulatory requirements throughout the intended period of operation.

As mentioned, the implemented processes and programmes for the safe management of radioactive waste comply with national and international principles for radioactive waste management, that is:

- waste avoidance and minimisation;
- waste reuse;
- reprocessing and recycling;
- waste conditioning; and
- waste storage and waste disposal.

The programme is aligned with the following requirements:

- Department of Mineral Resources and Energy (*Radioactive Waste Management Strategy and Policy of South Africa*) [239]
- The requirements from R.388 (*Safety Standards and Regulatory Practices (SSRP)*) [241]
- NIL-01 Variation 19 (*Koeberg Nuclear Installation Licence*) [286]
- IAEA GSR Part 3 (R*adiation Protection and Safety of Radiation Sources: International Basic Safety Standard*) [244]
- IAEA SSR-2/2 (*Safety of Nuclear Power Plants – Design*) [267]
- IAEA GSR Part 5 (*Predisposal Management of Radioactive Waste*) [245]

The programme is implemented in accordance with the Eskom policy, 32-227 (*Radiation Protection and the Safety of Radiation Sources Policy*) [88], 238-51 (*Spent Fuel Management Strategy*) [23], Eskom standard 32-226 (*Requirement and Rules for Radiation Protection and the Safety of Radiation Source*) [87], and 240-113228853 (*Koeberg Solid Radioactive Waste Management Plan*) [35]. Responsibilities and accountability for the radioactive waste management programme are defined in KAA-634 (*Responsibilities for the Radioactive Material Control Programme*) [142], 240-113228853 (*Koeberg Solid Radioactive Waste Management Plan for Koeberg Nuclear Power Station*) [35], KSA-048 (*Management of the Solid Radioactive Waste Programme*) [179], KSH-10 (*Functional Responsibilities for Radiation Protection at Koeberg Operating Unit*) [185], and KSH-012 (*Radiation Protection Standards and Expectations*) [187]. The implementation of the radioactive waste management programme is documented in KWH-S-033 (*Processing and Administration of*

Solid Radwaste) [199] and KWH-S-047 (*Implementation of the Radioactive Material Control Programme*) [205].

9.7.3.2 Waste Identification, Quantification, Characterisation, and Classification

Koeberg has developed and implemented adequate processes for identifying, quantifying, characterising, and classifying radioactive waste. These processes will remain adequate for LTO. The processes are consistent with national and international practices (that is, the national radioactive waste management policy and strategy document for the Republic of South Africa [239], NIL-01 Variation 19 [286], regulations on safety standards and regulatory practices [241], and the IAEA GSR Part 5 (*Predisposal Management of Radioactive Waste*) [245]). Radioactive waste is classified in accordance with the national radioactive waste classification scheme as contained in the radioactive waste management policy and strategy of South Africa [239].

Radioactive waste streams are identified, characterised, and classified to ensure:

- the identification and nature of site-specific radioactive waste streams or categories and associated waste management issues;
- consideration and listing of realistic options for the long-term management of specific radioactive waste management streams or categories;
- systematic evaluation of the merits and disadvantages of each option (multi-attribute analysis or any other suitable methodology covering cost-effectiveness, technological status, operational safety, and social and environmental factors);
- identification of the best available technology not entailing excessive cost (BATNEEC);
- acceptance of BATNEEC as a waste stream or category-specific strategy; and
- review mechanisms of industry- and site-specific waste management plans.

The adequacy and effectiveness of the waste identification, quantification, characterisation, and classification process were assessed during the PSR safety performance review. During the assessment, no deviations from national or international requirements were identified. The assessment concluded that Koeberg maintained records of radioactive waste generated and accumulated on site and that the process was based on regulatory requirements [67] as documented in 240-113228853 (*Koeberg Solid Radioactive Waste Management Plans*) [35], KSA-048 (*Management of the Solid Radioactive Waste Programme*) [179], and KWH-S-033 (*Processing and Administration of Solid Radwaste*) [199].

The solid radioactive waste management plan outlines the radioactive waste streams generated by the facility. Separate waste management plans need to be produced for specific projects such as the SGR replacement and submitted to the National Committee on Radioactive Waste Management (NCRWM) for review and approval. (Refer to § 9.7.3.10.) The waste management plans for future project waste streams will be developed during the project development stages and submitted to the appropriate authorities for review and acceptance.

9.7.3.3 Low- and Intermediate-Level Waste – Short-Lived (LILW-SL)

The control of the radioactive waste management programme is documented in KAA-634 (*Responsibilities for the Radioactive Material Control Programme*) [142]. Stored waste can be retrieved for clearance, processing, or disposal later. Before disposal, LILW-SL generated is stored on-site at the low-level waste building (LLWB) temporarily (see § 9.7.3.7). The waste is stored in concrete or steel drums before being transported to Vaalputs.

However, it is noted that, in recent years, a backlog in the shipping of a high volume of radioactive waste has been experienced. This was caused by changes in the waste acceptance criteria and other regulatory requirements that could not be met, which affected the shipment of waste to Vaalputs. The backlog shipments resulted in less storage space in the LLWB.

Subsequently, regulatory approval was obtained to resume shipment of specific waste packages. Shipping is being expedited to reduce the backlog and increase storage space in the LLWB. An action plan has been developed to address outstanding issues related to the shipment of all waste packages, and the actions are scheduled to be completed by 2024.

9.7.3.4 Processing of LILW-SL

The processing of radioactive waste includes its pre-treatment, treatment, and conditioning. Koeberg has adequate processes for LILW-SL, and the requirements for waste processing are documented in the following procedures:

- KWW-TES-003 (*Encapsulation of Radioactive Water Filters in Concrete Drums*) [207]
- KWW-TES-009 (*Compacting Low-Level Waste into 210-Litre Steel Drums*) [208]
- KWW-TES-011 (*Encapsulation of Radioactive Concentrates in Concrete Drums*) [210]
- KWW-TES-021 (*Encapsulation of Non-Compactable Radioactive Waste in Concrete Drums*) [212]
- KWW-TES-020 (*Encapsulation of Radioactive Sludge in Concrete Drums*) [211]
- KWW-TES-024 (*Drumming of Non-Compactable Radioactive Waste in Steel Drums*) [213]

Although LTO will result in additional waste generated during the 20-year period, the waste will be managed in line with the existing radioactive waste management programme, which has been demonstrated to be adequate for LTO.

9.7.3.5 Waste Reduction Programme

Koeberg has adequate processes and procedures for the minimisation and safe management of radioactive waste on-site. The waste reduction programme was assessed during the PSR safety performance review, and it was concluded that adequate measures to reduce waste existed and that the programme had been implemented satisfactorily. The processes and requirements used to minimise radioactive waste are set out in the following documents:

- 238-54 (*Radiation Protection Licensing Requirements for Koeberg Nuclear Power Station*) [26]
- 238-51 (*Radioactive Waste Management*) [23]
- 238-19 (*Generation Division Radiation Protection Manual*) [7]
- 32-227 (*Radiation Protection and Safety of Radiation Sources Policy*) [88]
- 240-113228853 (*Solid Radioactive Waste Management Plan for Koeberg Nuclear Power Station*) [35]
- KAA-634 (*Responsibilities for the Radioactive Material Control Programme*) [142]

Should there be any improvement or changes to the process (based on operating experience, technology improvements, or any other activity that may trigger the change), the appropriate change management processes will be followed to implement the changes.

9.7.3.6 Clearance, Authorised Discharge, Disposal, Reuse, or Recycling of Radioactive Waste

Radioactive material for which no further use is foreseen and with characteristics that make it unsuitable for authorised discharge, authorised use, or clearance from regulatory control is processed as radioactive waste. The steps leading to safe clearance, authorised discharge, disposal, reuse, or recycling are consistent with limiting the exposure from such activities and are documented in 240-113228853 (*Koeberg Solid Radioactive Waste Management Plans*) [35] and KSA-048 (*Management of the Solid Radioactive Waste Programme*) [179].

9.7.3.7 Low-Level Waste Storage Building

The LLWB was designed as a warehouse to store waste drums on site, and provision is made for regular monitoring, inspection, and maintenance of the waste and storage facility to ensure its continued integrity. A basic inspection is conducted annually and a comprehensive inspection every five years.

The LLWB stores waste such as active spent resin, evaporator concentrates, and miscellaneous waste. The classification of the radwaste stored in the LLWB and the radiological classification of the LLWB area will remain unchanged for the period of LTO. The dose rates are surveyed weekly in accordance with KAH-002 (*Radiation Surveillance Programme*) [159] and remain within limits. The occupational dose continues to be monitored and does not show an adverse trend. A project plan and the methodology (NSIP04207 (*Methodology for Documenting the Use of the LLWB*) [218]) for performing the assessment were submitted and approved by the NNR.

A safety assessment was performed to evaluate the adequacy of the LLWB to store waste. This assessment considered public, occupational, inspection, external hazard, structural and shielding requirements. Based on the assessment, it was concluded that the LLWB was safe for storing LILW-SL during the extended life of Koeberg and that the risks to the public and the environment would remain within the regulatory requirements.

The adequacy of the storage capacity is periodically reviewed, taking into account the predicted waste arising both from normal operation and possible incidents, the expected lifetime of the storage facility, and the availability of disposal options [67].

9.7.3.8 Transport of Radioactive Waste

Radioactive material is transported off site in accordance with the requirements of the IAEA's SSR-6 (*Regulations for the Transport of Radioactive Material*) [268]. The transport of radioactive waste is covered under § 9.69.6 of this document (impact of LTO on RP). An assessment was performed during the PSR safety performance review to evaluate the effectiveness of the transportation of radioactive material [67], and Koeberg complies with the NNR and IAEA safety standards for the safe transport of radioactive material. The responsibilities and accountability for the transport are documented in KAA 634 (*Responsibilities for the Radioactive Material Control Programme*) [142], KWH-S-033 (*Processing and Administration of Solid Radwaste*) [199], and KWH-S-037 (*Classification of Solid Radioactive Materials and the Acceptable On- and Off-Site Packaging Requirements for Such Materials*) [200]. The procedures are consistent with international good practice contained in the IAEA safety standards for the transport of radioactive material. These procedures will remain applicable during LTO.

The roles, responsibilities, and expectations for the transport of radioactive material are documented in KAA-634 (*Responsibilities for the Radioactive Material Control Programme*) [142], KSH-012 (*Radiation Protection Standards and Expectations Koeberg Procedures*) [187], and KSH-010 (*Functional Responsibilities for Radiation Protection at Koeberg Operating Unit*) [185]. In addition to RadPro and the radioactive waste (radwaste) tracking programme (RTP) databases, these documents are included in the management system for transporting radioactive material. RadPro controls access to controlled zones and records of radiation worker doses. The RTP is used to keep an inventory of radioactive waste on-site. Events involving the transport of radioactive material, whenever they occur, are raised on DevonWay and investigated.

9.7.3.9 Disposal of LILW-SL

The LILW-SL is disposed of in near-surface trenches at the Vaalputs' national radioactive waste disposal facility in the Northern Cape. Koeberg's volume of LILW-SL waste produced since the commissioning of the facility provides a basis for the waste volume projection for LTO. A total of 5600 concrete drums of waste, with a total volume of 11 000 m^3 was produced between 1984 and 2022. Therefore, a total of 3000 concrete drums, with a total volume of 5800 m^3 is projected for the LTO period. Similarly, the total number of steel drums produced between 1984 and 2022 was 27 400 drums with a total volume of 6200 m^3. A total of 16 500 steel drums with a total volume of 3750 m^3 is projected for the LTO period.

It was confirmed with the National Radioactive Waste Disposal Institute (NRWDI) that Vaalputs has adequate storage capacity to accommodate waste generated during the LTO period [219]. Procedures used to ensure that only dry, solid, and solidified or immobilised radioactive waste is shipped for disposal are as follows:

- KWW-TES-003 (*Encapsulation of Radioactive Water Filters in Concrete Drums*) [207]
- KWW-TES-009 (*Compacting Low-Level Waste into 210 Litre Steel Drums*) [208]
- KWW-TES-010 (*Final Capping of Concrete Drums*) [209]
- KWW-TES-011 (*Encapsulation of Radioactive Concentrates in Concrete Drums*) [210]
- KWW-TES-021 (*Encapsulation of Non-Compactable Radioactive Waste in Concrete Drums*) [212]
- KWW-TES-020 (*Encapsulation of Radioactive Sludge in Concrete Drums*) [211]
- KWW-TES-024 (*Drumming of Non-Compactable Waste in Steel Drums*) [213]

9.7.3.10 Waste Generated by Major Modifications

For major projects (for example, SG, PTR tank, and RPV head replacements), special attention is paid to the characteristics of the waste before a final decision is made on the ultimate radioactive waste management option. As specified in document 240-113228853 (*Koeberg Solid Radioactive Waste Management Plan*) [35], separate waste management plans will be submitted to the National Committee on Radioactive Waste Management. Currently, there are separate waste management plans for the following project waste streams:

- Original steam generators

  After removal from the reactor building, the original steam generators (OSGs) will be stored at the Koeberg site for an interim period. This will allow for all the transport and disposal approvals and for Vaalputs to be prepared to receive the OSGs for final disposal. The OSGs will be stored at the Koeberg site in compliance with NIL-44 Variation 1 (*Original Steam Generator Interim Storage Facility (OSGISF)*) [215]. In the meantime, the OSGs will be stored in the interim storage facility (known as the OSGISF). To this end, an OSGISF has been constructed at the Koeberg site. The radioactive waste management plan for the OSGs [54] has been conditionally approved by the. The NNR will be consulted at the different stages of the project before the final submission of the safety justification and revised post-closure radiological safety assessment. The secondary waste arising from SGR will be managed in accordance with 240-113228853 (*Koeberg Solid Radioactive Waste Management Plans*) [35].

- Refuelling water storage tanks (PTR tanks)

  The Koeberg PTR tanks have been replaced due to their degradation associated with atmospheric stress corrosion cracking (ASCC). The tanks are currently temporarily stored in the low-level waste holding area. The tanks will be decontaminated and cleared from regulatory control. The solid radioactive waste management plan has been updated to incorporate the management of residual waste arising from the dismantling and decontamination of the PTR tank. The updated solid radioactive waste management plan has been submitted to the NCRWM for approval.

- Reactor pressure vessel head (RPVH)

  The RPVH replacement project will include the disposal of the original Unit 1 RPVH, the original Unit 2 RPVH, and the original Unit 2 control rod drive mechanism (CRDM). The original RPVHs and control rod drive mechanisms are stored in the LLWB. Eskom will develop a radioactive waste management plan for the final disposal of the RPVHs and submit the plan to the NCRWM for approval. The date for the development and submission of the radioactive waste management plan will be communicated to the NNR.

In summary, project waste is adequately controlled, and it is not anticipated that significant volumes of waste will be generated due to additional major projects to support LTO.

9.7.3.11 High-Level Waste

Spent fuel is currently stored in two spent fuel pools and a number of dry storage casks on site. The spent fuel pool capacity is insufficient for LTO. This has necessitated the loading of spent fuel into dry storage casks.

A nuclear fuel strategy has been developed that caters for the transfer of spent nuclear fuel from the spent fuel pools into the dry storage casks. In addition, the utility is in the process of procuring spent fuel inserts that will be used to reduce the total reactivity and use the storage spaces that are currently unavailable in the spent fuel pools due to the checker-boarding arrangement of fuel assemblies. This will make currently unusable storage cells in the spent fuel pools available, allowing for an increase in the total number of spent fuel assemblies stored in the spent fuel pools.

In line with the strategy, Koeberg will store casks at the TISF, while the government is establishing the CISF. With regard to the establishment of the CISF, NRWDI informed Eskom that the CISF will be established by 2030. Eskom has applied for a nuclear installation licence (NIL) and environmental authorisation for the TISF. In 2017, Koeberg obtained environmental authorisation for the proposed TISF in terms of the National Environmental Management Act 107 of 1998 and the Environmental Impact Assessment Regulations, 2014. The granting of the NIL to construct and operate the TISF will serve as a mitigation to the unavailability of the CISF and will allow storage of spent nuclear fuel whilst the CISF is being established.

9.7.4 Impact of Long-Term Operation on the Environment

Koeberg's commitment to protecting the environment is set out in 32-727 (*Eskom Safety, Health, and Environmental Policy*) [89] and 32-227 (*Radiation Protection and the Safety of Radiation Sources Policy*) [88]. The environmental policy commits Eskom to zero harm to the environment, and the radiation protection policy ensures that nuclear and radiation safety receives the highest priority. An integrated management system ensures that the policy is properly implemented.

While this section focuses on the radiological impact on the environment, changes to the existing chemistry regime to address non-radiological environmental impacts will also be discussed, since these changes can affect plant reliability and radiological safety.

The section describes the potential impact of extended normal plant operations on the environment, which is shown to be insignificant. The sections below show that the effluent and environmental monitoring programmes ensure that emissions and discharges are properly controlled and are as low as reasonably achievable and that no changes to these programmes are required for LTO. This section also demonstrates, with input from the PSR [71], that operational practices to ensure environmental protection meet NNR requirements, follow relevant international guidelines, and ensure safe operation.

9.7.4.1 Impact of Changes to Plant Operations and Maintenance due to LTO

LTO introduces effects that have an impact on the environment during the period of extended operation. The identified and assessed risks to the environment include planned changes to the plant chemistry regime, plant refurbishment, plant ageing, change in radioactivity in the environment, and change in land use around the power plant. These changes are discussed and summarised below.

- **Change in chemistry regime**

  Koeberg has a seawater cooling water system to remove excess heat in the condenser. The seawater temperature increase from this process is restricted to approximately 10 °C to limit the impact on the environment. Given the restricted temperature increase, a high volume is necessary to ensure that the flow is large enough (around 164 000 kl/h for two units) for sufficient heat transfer. The intake seawater is chlorinated to prevent biofouling of Koeberg's seawater systems. When the seawater is discharged, this waste stream is used to co-discharge other effluents. These include radioactive effluent from plant processes, boron used for reactor power management, effluent from demineraliser plant, and chemicals used for corrosion control, such as lithium hydroxide, hydrazine ammonia and ethanolamine.

  In line with coastal environmental legislation [281], Koeberg is currently investigating the feasibility of reducing the discharge of chlorine and hydrazine to lower the adverse impact on the environment. Studies are in progress [74] to support decreased chlorination, and the results so far have shown that decreased chlorination will not adversely affect the biofouling prevention of the circulating water system (CRF). Studies are also in progress [60] to support the use of film-forming amines as an alternative to hydrazine for the layup of the auxiliary boiler system (XCA) and the secondary plant (including steam generators). Therefore, it is expected that, during LTO, the environmental impacts from chlorine and hydrazine will be reduced, while still adequately ensuring the protection of the plant systems against biofouling and corrosion, respectively. These changes will not adversely affect the indicator species used to monitor the impact and dispersion of radioactive effluents.

- **Plant refurbishment**

  Koeberg will undertake major refurbishment activities to ensure that LTO remains feasible, and that plant safety and reliability continue. Major refurbishments can have an impact on the environment, and the extent of the impact will depend on several factors. One of these factors involves replacing components exposed to the primary coolant. The material composition of the replacement components may differ from the original components, which may lead to a change in the composition of the corrosion products in the primary circuit and, hence, activation products found in effluent. Each replacement project of this kind has considered and will consider, changes in material composition to ensure the minimisation of radioactivity discharged and the impact on the annual authorised discharge quantities (AADQs).

  For example, due to the exposure of new material to the primary circuit, it was considered that initial operation with the new steam generators would at first result in higher reactor coolant system (RCP) activity and might result in changes in public exposure. The impact of this change was assessed as part of the SGR project [1], and the study concluded that the potential increase in effluent and the public dose was not significant. After the initial period of operation, reactor coolant system activity was expected to decrease. This SGR assessment, which used conservative assumptions, was performed using the current dose conversion factors (DCFs) and AADQs. Given that updated DCFs [217] and AADQs [79] are planned for implementation in the next 18 months and may coincide with the effects of the SGR, an assessment using the updated DCFs and AADQs was performed, and the results are shown in Table 9-6 below. The assessment demonstrated that the short-term increase in effluent activity had sufficient margin to the AADQ and that the contribution to public dose remained insignificant.

Table 9-6: *Potential Temporary Increase in Public Dose and Percentage AADQ After SGR Using Updated DCFs and AADQs Using Conservative Assumptions*

	Before SGR	Temporary Increase After SGR	Before SGR	Temporary Increase After SGR
	% AADQ	**% AADQ**	**Dose µSv**	**Dose µSv**
Liquid				
Co-58	7,1	20,5	0,007	0,02
Co-60	13,3	38,4	0,06	0,17
Gaseous				
Co-58	0,05	0,1	2,9E-04	8,4E-04

Replacement of plant components can also result in higher volumes of effluent. However, the higher waste volumes are temporary and will be minimised in line with the ALARA principle.

Furthermore, regarding non-radiological environmental impacts, the replacement of large components can have an impact on spatial areas around the plant. This is applicable when these replacements result in the need for large laydown areas, new equipment storage or fabrication facilities, and the temporary construction of large cranes required to move components. All of these activities are screened for potential impact on the environment. To date, two projects have triggered the need for an environmental authorisation in terms of the regulations of the *National Environmental Management Act 107 of 1998* [280]. The projects are, firstly, the extension of the car park to accommodate a larger workforce needed for large component change-out activities and, secondly, the replacement of the high-voltage yard at a new location outside of Access Control Point 2 (yet to commence). The car park extension did not have any radiological or plant safety impacts, so there was no need for NNR approval. Some aspects of the high-voltage yard replacement may require NNR approval due to the potential impact on electrical power supplies and their contribution to plant safety.

As discussed above, replacing large components in the RCP system can change the radioactivity usually found in the effluents discharged. Still, the overall impact has prospectively been shown to be insignificant. Non-radiological impacts are screened to determine whether environmental authorisations are required and are processed accordingly.

- **LTO impact of plant ageing**

  The SSCs used in waste treatment and effluent monitoring systems are in-scope SSCs important to safety. Therefore, the relevant ageing and degradation mechanisms will be managed according to their ageing management programmes.

  In the PSR radiological impact on the environment review, the actual plant condition of the effluent treatment systems was assessed. (See Appendix A of 240-161609494 (*KNPS 3rd Periodic Safety Review Report, Safety Factor 14: Radiological Impact on the Environment*)

[71].) The assessment concluded that the only plant condition risk related to normal and planned effluent discharges is the degradation of the liquid waste treatment system (TEU) evaporators, leading to a low radioactivity decontamination factor. Despite the deviation raised in the PSR, the current system performance and plant condition remain adequate. The downstream demineraliser 9 TEU 003 DE has a relatively high decontamination factor to compensate for any degraded performance of the evaporator, provided the demineraliser is replaced timeously. There is no evidence from the decontamination factor trends that evaporator degradation has changed much over the past 10 years. Performance monitoring of the evaporator will continue during the LTO period.

The civil structures form a barrier between radioactive substances and the environment. For example, cracks in the bund walls around the nuclear island liquid waste monitoring and discharge system (KER) or reactor cavity and spent fuel pit cooling system (PTR) tanks are known internationally to cause groundwater contamination after spurious tank overfilling events. Should minor unplanned leaks due to cracks in civil structures occur, the public dose impact will not be significant because the water movement is towards the sea, and there is no credible dose pathway. Since this is not a regulated discharge pathway, prevention and mitigation in monitoring and planned maintenance activities are required. These prevention and mitigation activities have been instituted in the updates to civil monitoring programmes to ensure that the civil structures are adequately maintained. For example, actions are taken to ensure that the SFP monitoring channels are clear of debris and that sumps have adequate waterproof coatings.

- **Change in radioactivity in the environment**

  Radioactivity can be found in the natural environment from radionuclides having a longer half-life. Taking radioactive decay into account, equilibrium in the radioactivity in the environment is reached before 40 years of operation for radionuclides with half-lives less than 10 years.

  For LTO, the only important radionuclides with half-lives longer than 10 years and with high bioaccumulation in plants and animals that could pose additional risk due to LTO are carbon-14 (a half-life of 5 730 years), strontium-90 (a half-life of 29 years), cesium-137 (a half-life of 30 years), and nickel-63 (a half-life of 96 years). Table 9-7 below shows that the increase in radioactivity in the marine environment is minimal for LTO. The increase in radioactivity in the terrestrial environment is not detectible.

Table 9-7: *Estimated Increase in Radioactivity in the Marine Environment of 60-year Compared to 40-year Operation for Important Nuclides with Long Half-lives*

Radionuclide	% Increase in Sea Sediment	% Increase in Crustaceans and Fish	% Increase in Molluscs
C-14	1,2	0,0	0,0

Radionuclide	% Increase in Sea Sediment	% Increase in Crustaceans and Fish	% Increase in Molluscs
Cs-137	2,3	0,0	0,0
Ni-63	5,8	0,3	0,0
Sr-90	1,3	0,0	0,0

The public dose was calculated considering the environmental build-up of 60 years for LTO operation in the radiological 'Environmental Surveillance Requirements' (238-47) [20] programme. The dose was estimated to be 94 µSv/a, which is below the dose constraint of 250 µSv/a.

The radiological environmental surveillance programme [20] tracks the radionuclide concentration in environmental samples and trends any significant environmental build-up for normal operations. Should any significant build-up occur, the radiological surveillance programme will show changes in trends during the LTO period.

Studies have been performed to determine the dose impact on plants and animals. The DSSR [216] assessed the dose impact of the build-up of radionuclides for 60 years, revealing that the dose to reference plants and animals is below the dose screening value of 40/400 µGy/h of the IAEA and UNSCEAR (United Nations Special Committee on the Effects of Atomic Radiation).

- **LTO impact on land use around Koeberg**

The environmental monitoring programme at Koeberg [20] is aimed at monitoring all important exposure pathways. These pathways can change from time to time, depending on changes in human activity around the power station and following important modifications made to the plant. According to regulatory requirements, an annual land-use review is performed within 10 km of the power station. The objective of the land survey is to identify new land uses, changes in receptor locations, or new routes of exposure.

Considering the PSR period of review, no new sources or pathways were highlighted, requiring a sampling location to be assessed. Agricultural activity around the power station is relatively static, given the poor quality of land for agriculture around the power station and the restrictions on development in terms of the emergency plan.

The possibility of a large desalination plant in the vicinity of Koeberg was assessed and was found not to be an important land-use change from a public dose perspective.

Any potential changes will be noted during the annual review, and if deemed necessary, their impact will be assessed. No new developments are foreseen to introduce new exposure pathways during the LTO period.

9.7.4.2 Plant Design Provisions for Environment Protection

Koeberg has a large dry containment structure that prevents the release of radionuclides into the environment in the event of an accident. Residual low levels of direct radiation emissions from the containment structure will also occur, but these are not detectible at the site boundary. However, the normal operation of the power plant does require the release of some radioactive effluent.

A large inventory of fission and activation products is built up in the fuel pellets during normal power operation of the reactor. Almost all fission and activation products are contained within the fuel pellets, and these fuel pellets are enclosed in metal cladding rods. Although most radioactivity is contained within the fuel pellets and cladding, a small fraction of the radioactivity escapes the fuel rods and contaminates the reactor coolant. The noble gas that has escaped from the fuel since 2010 is below 1% of the AADQs.

Apart from being below 1% of the AADQs, the trend of noble gases has also decreased since 2010, as shown in Figure 9-6 below. The reduction in noble gases is due to ongoing fuel cladding reliability improvements (design and manufacturing). LTO will have no adverse impact on the cladding reliability; therefore, noble gases will remain at a relatively low level.

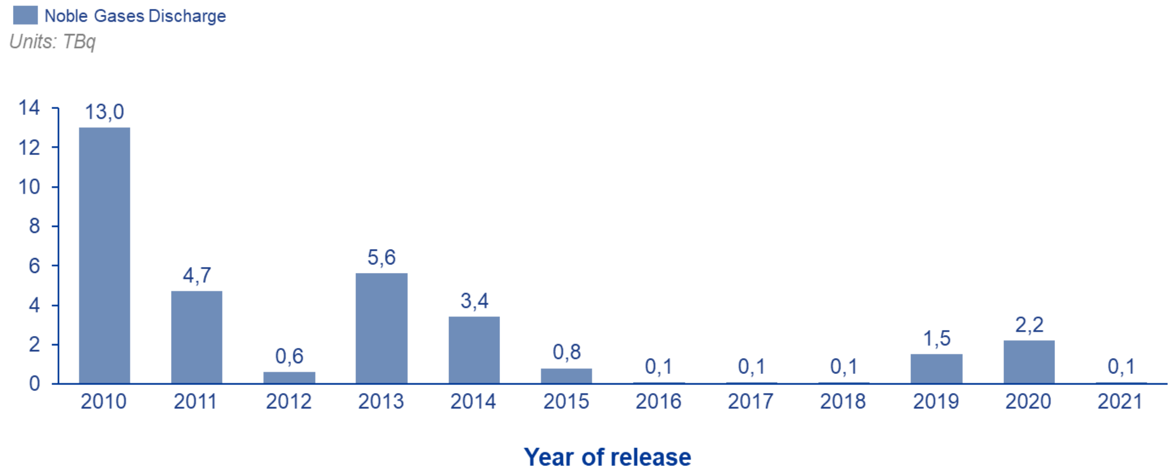

Figure 9-6: *Noble Gases Discharge From 2010 to 2021 (TBq)*

The radioactivity in the reactor coolant is the main source of gaseous, liquid, and solid radioactive wastes from the treatment of liquid and gaseous effluent. Apart from the radioactivity from the fuel, the primary system coolant also has radioactive contaminants from neutron activation. As discussed above, LTO can result in a temporary increase in the activated corrosion products in effluent when new material is introduced by replacing parts of the primary circuit, such as new steam generators. The increase is expected to be short-lived, and potential increases are assessed to determine the impact on dose as part of the plant design change process.

The design capability of the waste treatment systems and the possible effects of LTO are discussed below.

- Gaseous effluent management system design

  There are three primary sources of gaseous radioactive emissions:

  * Discharges from the gaseous waste management systems
  * Discharges associated with the discharge of non-condensable gases at the main condenser from primary-to-secondary system leaks or tritium diffusion across steam generator tubes
  * Discharges of radioactive gases from the building ventilation exhaust, including the reactor, auxiliary, and fuel-handling buildings

  The gaseous waste management system (TEG) collects fission and activation products that accumulate in the primary circuit. A small portion of the primary coolant flow is continually diverted to the reactor chemical and volume control (RCV) system to remove contaminants and adjust the coolant chemistry and volume. During this process, non-condensable gases are stripped and routed to the TEG system, consisting of two gas storage tanks. The storage tanks allow the short half-life radioactive gases to decay if time allows, leaving only relatively small quantities of long half-life radionuclides released into the atmosphere. Although a charcoal delay system was provided in the original design of TEG, this was never commissioned due to concerns related to effective monitoring of the subsystem. The PSR radiological impact on the environment review raised a deviation concerning the lack of TEG delay beds, which do not always ensure that radioactivity in effluent discharged is optimised. (A PSR deviation with a “low” safety grading was raised due to the low dose impact.) The PSR recommended that an assessment be performed to determine the suitability of, and need for, utilising the TEG delay beds. The outcome of the assessment will determine whether a reduction of the radioactive effluent discharged is required and, particularly, noble gases. Since LTO is not expected to increase the noble gas discharged given the expected continued improvements in fuel cladding reliability, this change will not be important for LTO. Nevertheless, the need for a TEG delay bed will be assessed in terms of ALARA.

- Liquid radioactive effluent management system design capability

  There are three sources of liquid effluent as a result of operations. The first is effluent streams coming from normally contaminated systems such as the reactor cooling system (RCP), the chemical volume and control system (RCV), the reactor residual heat removal system (RRA), and the spent fuel pit cooling system (PTR). The second effluent stream is from systems that are not normally contaminated but can become contaminated due to leakage across a barrier, such as from the component cooling system (RRI), the auxiliary steam distribution system (SVA), secondary systems (supplying steam to the turbine and feedwater to the steam generators), and the conventional island liquid waste monitoring and distribution system (SEK). The third type of effluent is from systems with more than two barriers, and contamination is extremely unlikely, such as the essential cooling water system (SEC) and circulating cooling water system (CRF).

Radioactivity in the primary system is the source of liquid radioactive effluent. This radioactivity will contaminate various systems, and the contaminated effluent is managed by systems such as the vents and drains system (RPE), the liquid effluent treatment system (TEU), the nuclear island liquid waste monitoring and discharge system (KER), the boron recycle system (TEP), and the laundries and decontamination workshop effluent system (SBE). Monitoring and discharge of the conventional island liquid waste system (SEK) manage secondary effluent, but SEK can also accept waste from KER. The TEU system segregates waste into process drains, floor drains, chemical drains (including clean condensate from the nuclear auxiliary building ventilation system (DVN)), and service drains (including laundry water). The TEP system is used for the deboration process.

The higher active effluent streams are treated either by ion exchange (RCV, TEP) or evaporation and ion exchange (TEU process and floor drains). The average activity in the primary circuit is about 6,1E8 Bq/m^3 during power operations, and 2,7E9 Bq/m^3 during RRA operations is reduced after effluent treatment to an annual average between 1,1E5 Bq/m^3 and 5,1E5 Bq/m^3 found in KER (2014 to 2021 data). The radioactivity of service drains (around 6,5E4 Bq/m^3 in 2018) and chemical drains (around 3,4E6 Bq/m^3 in 2018) is usually low (less than 1E6 Bq/m^3) and does not require treatment. At times, the chemical drain tanks may have higher radioactivity (up to 1E7 Bq/m^3), but this is when floor drains become chemically polluted and are routed to the chemical drains tank (for example, when RRI is drained inside the containment). The APG has an ion exchange system to ensure good secondary chemistry and removes some radioactivity from the secondary system.

Increased effluent treatment may reduce the radioactivity discharged; however, it may increase the volume of solid waste that requires disposal. The extent and types of treatment depend on the chemical and radionuclide content of the effluent. A trade-off is sometimes required since the evaporators can produce a relatively high volume of solid waste due to the boron concentration in the waste systems.

The designs and ageing of the liquid waste treatment systems were reviewed against NNR and international requirements in the PSR radiological impact on the environment assessment. While the PSR found the design of the waste treatment systems to be adequate, the design did not always ensure that radioactivity in effluent discharged was optimised. (A PSR deviation with a “low” grading was raised.) Recommendations were made to assess the suitability of and need for, the treatment of the remaining water in the boron recycle system (TEP) after an outage, the need to bypass floor drains when draining the component cooling system (RRI) in containment, and a solution to the high solid waste generated by the TEU evaporators. The

PSR recommended that alternative design or compensatory actions be sought and assessed to optimise radioactivity discharge. These recommendations have been included in the PSR IIP. Therefore, the design of gaseous and liquid waste management systems is suitable for continued operations, and radioactive release monitoring will continue during LTO according to the effluent monitoring programme. Solid radioactive waste is not released into the environment at Koeberg, and the design of solid radioactive waste management plant systems is discussed in § 9.7.3.

- Effluent monitoring system design

  The radiation monitoring system (KRT) monitors the discharge of radioactivity released into the environment. If the radioactivity exceeds predetermined thresholds, an alarm or automatic protection actions are implemented, such as termination or diversion of the release.

  The KRT system performs a significant function in continuously monitoring all major and potential radioactive release paths to personnel and the public during normal operations and postulated accidents. The KRT system, thus, protects the environment should unusual discharges occur.

  The KRT system was replaced strategically over a planned period in various plant locations, starting in October 2012 and completed in October 2015. The system was replaced to ensure that reliability would continue unaffected, to address obsolescence, and to gain confidence that the system functionality of continuous monitoring would be achieved. The design of the new system provides approximately 20% more channels for monitoring of potential sources (or levels) of radioactivity. Each steam generator is monitored by a KRT channel for primary to secondary leak rate and gamma activity. The blowdown water from each steam generator is also monitored by individual KRT channels. The system has the capability to be expanded and upgraded for future modifications.

  The PSR [71], Appendix A, section 4.2 confirmed that the radiation monitoring system (KRT) instrumentation was suitable for responding to unplanned releases and was suitably designed and available. The KRT system is included in the scope of the SSCs important to safety subjected to ageing management review. This will ensure that ageing or degradation mechanisms will be managed to ensure system reliability during the LTO period. The KRT system is adequate for LTO, as it has been manufactured with modern technology, adequate spares are available for the foreseeable future, and obsolescence is not currently a risk.

9.7.4.3 AADQs and Effluent Discharge Conditions

The operational limits for discharges (known as AADQs) and effluent discharge conditions (described in the 'Liquid and Gaseous Effluent Management Requirements for KNPS' (238-49) programme [22]) have been developed by Eskom and approved by the NNR. The AADQs ensure compliance with the discharge dose limits in RD-0022 (*Radiation Dose Limitation at Koeberg Nuclear*

Power Station) [288]. The current dose assessment in the SAR [178], based on the current activity migration model (AMM), shows that the established dose constraint [241], [288], and the requirements of the IAEA Safety Standard Series GSR Part 3 (*Radiation Protection and Safety of Radiation Sources*) [258] are met.

The PSR Appendix A [71] concluded that the effluent discharge limits are generally well established and in line with NNR and international practice. Two PSR deviations were raised: one with a "low" safety significance related to the lack of inclusion of three important radionuclides from the programme (iron-55, nickel-63, and carbon-14) and the other graded as "drop" related to the lack of optimisation of the AADQs. The radionuclides that are not included in the programme are not being measured and are not included in the quarterly and annual retrospective dose assessment. These radionuclides are known as hard-to-detect nuclides, one of the main reasons for their exclusion in the past. The PSR recommendation that these radionuclides be included in the effluent monitoring programme has been listed in the PSR IIP and will be implemented in accordance with the schedule.

The "low" graded deviation mentioned above was not graded as a significant issue (that is, "low" grading), mainly because the public dose contribution of iron-55, nickel-63, and carbon-14 is expected to be less than 10 µSv/a, which is not seen as significant compared to the dose limit of 1 000 µSv/a. However, it is important to include these principal nuclides in the effluent monitoring programme and the public dose assessment.

Over the past 10 years, plant operating performance has shown that dose contribution from effluent discharges is generally a small percentage of the AADQ. This is expected to continue during LTO.

9.7.4.4 Normal Operations Environmental and Effluent Monitoring Programmes

Using effluent data and environmental concentrations over the past decade, the PSR radiological impact on the environment review confirmed that the annual estimated public dose had been well below 1% of the 1 mSv public dose limit (Appendix A, section 8). This estimated dose was well below background radiation levels.

Although the programme is implemented as required and in line with international requirements, six deviations were identified and graded as insignificant (ranging from "drop" to "low"). The deviations related to the inadequate trends and analysis of trends, ESL detector performance issues (issue since resolved), insufficient programme review, some procedural deficiencies, inadequate on-site groundwater monitoring, and inadequate reporting. The resolution of these deficiencies is addressed in the PSR IIP. Although the LTO will not significantly increase the planned discharge of radioactivity to the environment, an update to the effluent and off-site environmental monitoring programmes in line with the PSR IIP will be implemented during the LTO period in accordance with regulatory requirements and to ensure ALARA.

Appendix C of the PSR [71] found that several leaks and spills had been reported at Koeberg in the past 10 years (eight in total), and these had been reported to the NNR as required. The leaks or spills were minor in terms of activity released and did not pose any environmental concerns.

International operating experience shows that unexpected contamination of on-site groundwater can be a common challenge for many power plants due to above-ground spills and below-ground leaks. As a result, industry-led initiatives have been implemented by nuclear power plants internationally to protect the groundwater from contamination [283]. Ageing of the power station is an important contributor to the risks due to degradation of piping, coatings, sumps, and civil infrastructure [249], as discussed in § 9.7.4.1.

Considering these experiences, the ageing of systems could lead to undetected leaks and spills. Appendix B of the PSR [71] determined that the on-site monitoring to ensure a high probability of the prompt detection of a release of new sources of radioactive contamination to the environment from a leak or spill is inadequate (deviation graded "low"). This is discussed and assessed below.

Monitoring of on-site groundwater for radioactivity was started at Koeberg in 2001, and low contamination levels were found in the groundwater. In line with international norms, a site conceptual model was developed for Koeberg in 2016 to better understand the groundwater flow and ascertain where and how to design monitoring boreholes. The study found that the existing monitoring boreholes were inadequate to detect contamination on site. They were not always in the correct location and did not allow sampling at the correct depth. This indicates that the current measurement results add little value and require an interrogation of the relatively low values of measured results.

Over the past few decades, international operating experience related to the impact of leaks and spills on groundwater contamination has been shared with Koeberg. While many groundwater protection initiatives shared by EDF have been implemented at Koeberg, an effective groundwater monitoring programme has not yet implemented. Eskom has developed a groundwater monitoring programme [50] and is ready for implementation in accordance with the PSR IIP.

The Institute of Nuclear Power Operations (INPO) of the United States groundwater protection initiatives concluded, "In several cases, station groundwater protection programmes were not sufficiently rigorous to prevent and detect unexpected tritium releases to the environment outside of licensed effluent pathways". This statement is also applicable to Koeberg since the suggested improvements have not been fully implemented. Given that there is no groundwater pathway to the public, this is not a significant issue (the related deviation is graded "low"); nevertheless, Koeberg will implement an improved groundwater monitoring programme as indicated in the PSR IIP to ensure the prompt identification and characterisation of underground contamination issues.

The potential impacts from LTO that were identified and assessed included planned changes to the plant chemistry regime, plant refurbishments, ageing of the plant, change in radioactivity in the environment, and changes in land use around the power plant. These changes were assessed and were found not to be significant.

The PSR [71] confirmed that the environmental impact of the plant is insignificant compared to other radiation sources. The measures to control and monitor effluent discharges to the environment are appropriate and meet NNR and international expectations. LTO can safely continue based on the PSR conclusions and the findings of the assessment of the impact of LTO on the environment.

9.7.5 Organisational Provisions for Long-Term Operation

9.7.5.1 Organisational Structure and Policies

The organisational structure, management systems, and policies of the NOU are deemed adequate and meet the requirements of RD-0034 (*Quality and Safety Management Requirements for Nuclear Installations*) [290].

According to the NOU integrated management system 238-8 (*Nuclear Safety and Quality Management Manual*) [28] for all operations, the organisational structure has been compiled with due consideration given to safety management throughout the life cycle of the plant.

Document 32-83 (*Nuclear Management Policy*) [90] provides the framework for the life cycle of the plant and is managed through an operational plan, with a specific focus on stabilising, sustaining, and growing operations. The operational plan is reviewed annually.

Documents 32-83 (*Nuclear Management Policy*) [90] and 238-8 (*Nuclear Safety and Quality Management Manual*) [28] were reviewed against national and international requirements during the PSR to confirm their adequacy for the safe operation of the plant for LTO. The review concluded that the current policy and management system are adequate for plant operation and that the policy met the requirements of RD-0034 (*Quality and Safety Management Requirements for Nuclear Installations*) [290].

The organisational structure of the NOU as described in 240-64602879 (*Nuclear Operating Unit Structure and Mandates*) [81] contains details of roles and responsibilities of all functional areas for the period of LTO, and the Nuclear Engineering Department has been tasked with control of ageing management and pursuit of LTO.

Nuclear Engineering, as the nuclear design authority, and the Nuclear Engineering centre of excellence at Eskom have put arrangements in place for ageing management throughout the life of the plant, from current operations into LTO, and including decommissioning. One of the functions of the Nuclear Engineering Department is to provide, maintain, and manage engineering programmes and the ageing management process to ensure that the material and equipment integrity of systems, structures, and components is maintained in a safe, reliable, and functional state until the end of plant life.

9.7.5.2 Arrangements for Human Resources

Document 240-156938857 (*NOU Human Resources Position Strategy for Long-Term Operation*) [56] describes human resource processes and procedures to support LTO. Document 240-123782330 (*NOU Workforce Plan*) [38] provides the workforce plan for the next 10 years and is reviewed annually. The workforce plan was developed to meet Koeberg's resource needs, as informed by the operational plan and historical trends, such as attrition, an ageing workforce, and pipeline requirements. The strategic resource plan considers using a combination of permanent resources, supplemented by fixed-term contractors, and service contracts.

Over the past 39 years, the human resource planning process has provided adequate human resources for safe operations. The use of the pipelining process in technical departments, such as Maintenance, Operation, and Engineering, has ensured that staff vacancies are replenished. In 2015, the long-term planning for reactor operators resulted in the appointment of 100 additional reactor operator trainees.

However, due to global industry activities, in recent years, Koeberg has experienced higher-than-normal rates of attrition. The shortfall has been balanced by supplemental workers, through outsourcing of services (consultants and fixed-term contractors). RD-0034 is complied with to ensure compliance with the quality requirements for outsourced services and products. Additionally, where a skills and expertise shortfall exist, Koeberg has long-term partnerships with original equipment manufacturers, service providers, and other utilities (such as EDF) to provide technical support to compensate for the shortfall. Koeberg's resource provisions have always been supported by the Eskom executive management, especially resources to support safe operations of the plant.

Permanent cessation due to expiry or termination of the operating license is covered by the decommissioning strategy and decommissioning plan. It is not Eskom's intention to temporarily suspend operations (temporary cessation) at Koeberg. The arguments in the safety case pertaining to the availability of sufficient resources and technical support for the LTO period implicitly include periods of temporary cessation of operations. Business as usual will be applicable during temporary cessation. However, it is acknowledged that for temporary cessation of operations for extended periods (for example, exceeding 12 months), a safety assessment considering the aspects discussed in IAEA SRS No. 31 (*Managing the Early Termination of Operation of Nuclear Power Plants*) [254] will be compiled if such temporary cessation of operation is decided/required.

In the PSR human factors review, it was concluded that the current human resource planning processes are adequate and would remain adequate to manage resource requirements for safe operations in the LTO period [70].

In anticipation of LTO, Eskom has embarked on a recruitment campaign to fill vacancies (employment of permanent staff) to ensure that there are adequate resources for the LTO period. Document 240-156938857 (*NOU Human Resources Position Strategy on Long-Term Operation*) [56] contains evidence that demonstrates adequate arrangements for human resources.

9.7.5.3 Organisational Oversight Structures to support Ageing Management

In line with good practices relating to equipment reliability, the plant has established various forums to support the implementation of ageing management activities. It has developed a set of key performance indicators (KPIs) in line with industry practices to monitor the implementation of these activities and for continuous improvement purposes.

- Forums include the Plant Health Committee, programmes oversight committees (POC), all the work management forums for work preparation and execution, the Engineering Change Management Committee, and the Modification Review Committee for managing plant changes, etc.
- KPIs include work management key performance indicators, plant health indicators, programme health indicators, and safety systems availability.

The forums and indicators are based on the WANO performance criteria and objectives framework.

9.7.5.4 Competency Management

To manage employee competency for ageing management, the NOU has appropriate training arrangements and processes to ensure that Koeberg has a skilled and knowledgeable workforce in sufficient numbers, both now and into the LTO period.

Competency management aims to ensure that Koeberg will continue to have the requisite skills and expertise for safe plant operations. A sufficiently skilled workforce is achieved by establishing training and qualification programmes to ensure that adequate personnel are trained, qualified, and deemed competent to accomplish assigned duties. Specific qualification requirements are established for critical and unique job categories requiring specialised nuclear-related technical skills.

The operator training programme is internationally accredited and is regularly reviewed by international oversight bodies. The generic training programme was developed according to national and international requirements in accordance with ANSI 3.5 1998 (*Nuclear Power Plant Simulators for Use in Operator Training and Examination*) [230], the NNR licence document LD-1093 (*Requirements for the Full Scope Operator Training Simulator at Koeberg Nuclear Power Station*) [285], and others.

The NOU has a management training programme for professional development, leadership, management development, and soft skills for employees in leadership positions.

The skills required to practise ageing management and LTO activities are managed within the various line functions. Suitably skilled persons are available to support the requirements of an ageing plant. The NOU has comprehensive training programmes for the various departments. The main department tasked with ageing management is the Engineering Programmes Department, and the required training is managed in accordance with Document 331-148 (*Programme Engineer's Guide*) [97]. The training requirements for maintenance and inspection programme ageing management activities are documented in 240-149139512 (*Ageing Management Requirements for KNPS*) [52] and managed in accordance with the Nuclear Operating Unit management system.

The PSR human factors review concluded that human factors at Koeberg are well managed and documented in the Koeberg processes, procedures, and guidelines [70]. Some "low" graded deviations were identified, and corrective actions to address the deviations were developed and are included in the PSR IIP. The deviations will be resolved commensurate with the risk and managed within the context of continuous improvement; this will ensure that the NOU is well placed to prevent any significant human factors that can adversely affect continued safe operation.

9.7.5.5 Knowledge Management

Eskom endeavours to ensure that knowledge management is implemented throughout Koeberg. A comprehensive assessment of nuclear knowledge management requirements and processes was conducted in accordance with IAEA requirements. The assessment is documented in 240-106374672 (*Koeberg Pre-SALTO Self-Assessment Report*) [31]. While Koeberg had most of the elements of an effective knowledge management programme at its disposal, certain areas were found deficient, detracting from the coherence required from a goal-oriented knowledge management programme.

An effective knowledge management programme constitutes an integrated, systematic approach to identifying, managing, and sharing the knowledge of an organisation and enabling employees and staff to collectively create new knowledge to help achieve the objectives of the organisation.

The NOU decided to establish knowledge management as a key organisational programme to identify, manage, and share the knowledge within the NOU and assigned the function of piloting the implementation and integration of knowledge management at the NOU to Nuclear Engineering. The objective is to implement a knowledge management programme within the Nuclear Engineering Department (NE) and transfer the learning and experience from this pilot programme to other areas of the NOU. The Nuclear Engineering Department has developed document 240-146686589 (*Knowledge Management Standard*) [49], which contains the knowledge management process-related documents and repositories. The knowledge management processes use an integrated approach to identifying, capturing, evaluating, retrieving, and sharing relevant Koeberg information assets (such as databases, documents, policies, procedures, and previously uncaptured expertise and experiences from individual workers). The objective is to minimise the risk of losing critical knowledge necessary for safe operation and achieving Koeberg's main business objectives. The knowledge management programme also interfaces with other human resource processes such as succession planning, talent management, training, and job shadowing.

The 2021 WANO peer review found the nascent knowledge management pilot programme within Nuclear Engineering to be a strength. It recommended that the process be implemented in all departments within the NOU. Details of the implementation are contained in the LTO IIP. (Refer to Appendix A.)

9.7.5.6 Arrangements for Financial Resources

Adequate financial resources are available to support the performance of safety-related activities. Major expenditure for Koeberg is associated with salaries (operational cost) and the safety improvements in the nuclear technical plan (capital costs). The LTO integrated preparation plan (pre-LTO activities) has been sufficiently resourced and funded. During the LTO period, financial needs are driven by the LTO implementation plan. This plan is made up mainly of the PSR IIP. Therefore, in the first 10 years of the LTO period, the financial provisions will be primarily to support the execution of the PSR IIP.

A skills, time, and cost analysis was conducted for the PSR IIP (refer to Appendix I of document 331-608 (*KNPS 3rd Periodic Safety Review Global Assessment Report and Integrated Implementation Plan Report*) [115]) and concluded that the cost requirements for the PSR IIP were in line with past approved expenditure for a similar scope of activities. It can be demonstrated that Eskom has adequately managed its operational costs over the past 39 years. Therefore, financial resources for LTO are available for operational costs and the LTO scope of activities.

9.7.5.7 Arrangements for Tools and Equipment

The arrangements for tools and equipment to support operations (including outages, modifications, maintenance, etc.) will continue to follow current commercial, operating, chemistry, maintenance, and engineering processes. These processes are adequate for the timeous sourcing of tools to ensure availability throughout the life of the plant.

Eskom has ensured that the necessary materials are available for ongoing operations. Ageing management and LTO activities are currently funded, including securing original equipment manufacturer (OEM) services for inspections, modifications, studies, and other projects related to LTO.

The procurement of tools and equipment is undertaken in accordance with 32-1034 (*Eskom Procurement and Supply Chain Management Procedure*) [86].

9.7.5.8 External Organisations

Partnerships with organisations such as WANO, EDF, and EPRI are some of the tools available for alignment with the nuclear industry.

Throughout the life of the facility, Eskom has maintained a successful and mutually beneficial relationship with industry experts. Due to the long-term nature of these partnerships, they remain applicable during LTO. Arrangements have been made with the external organisations listed below to support Eskom with ageing management (among other requirements) for the period of LTO.

- EDF (Électricité de France)

Koeberg has established and continues a partnering contract with EDF for information sharing, benchmarking, and technical support since the plant design is similar to the EDF CPY fleet.

- EPRI (Electrical Power Research Institute)

  EPRI is an organisation that conducts research and development related to the generation, delivery, and use of electricity to help address challenges in electricity, including reliability, efficiency, affordability, health, safety, and the environment. Eskom benefits from research findings through its relationship with EPRI.

- WANO (World Association of Nuclear Operators)

  Eskom is a member of WANO-Atlanta, an organisation whose aim is to maximise the safety and reliability of nuclear power plants worldwide by working together to assess, benchmark, and improve performance through mutual support, exchange of information, and emulation of good practices.

- IAEA

  The IAEA is the international centre for co-operation in the nuclear field. The IAEA promotes the safe, secure, and peaceful use of nuclear technologies. The IAEA has been assisting Koeberg in the development of the LTO programme by providing guidance and peer reviews in LTO programme deliverables, such as the PSR, SALTO and others.

9.7.5.9 Procurement and Supplier Management

Eskom's procurement and supplier management procedures and processes meet national and international requirements and are adequate for current and continued operations.

Eskom's procurement documents were assessed against the requirements of RD-0034 (*Quality and Safety Management Requirements for Nuclear Installations*) [290] and international requirements in PSR organisation, management system, and safety culture review and found to be adequate for LTO. Processes related to supplier management are contained in the following procedures:

- 238-101 (*Quality and Safety Management Requirements for Nuclear Supplier Level 1*) [5]
- 238-102 (*Quality and Safety Management Requirements for Nuclear Supplier Level 2*) [6]
- 238-219 (*Level 1 Supplier SCEP Requirements*) [8]

9.8 Incorporation of Operating Experience for Continuous Improvement of Plant Safety

Koeberg utilises operating experience (OE) to improve the safety of the plant and safety-related programmes. OE is embedded in plant safety-related activities and processes. The fundamental basis of the performance improvement programme is the learning process through operating experience. This section describes the adequacy of the performance improvement programme in support of safe LTO.

The facility has a mature OE programme that ensures that the organisation benefits from internal and external industry experience. The programme is structured such that OE is used to prevent or mitigate adverse events and improve the safety of the plant, the workers, and the environment.

The operating experience and continuous performance improvement elements were assessed during the PSR in multiple safety factors. The OE and performance improvement programmes were assessed in PSR safety performance [69] and international operating experience reviews [68]. The review concluded that the OE programme at Koeberg is robust and comprehensive, with no deviations against national and international requirements. Furthermore, the PSR concluded that:

- the OE programme at Koeberg currently meets NNR, WENRA, and IAEA requirements;
- the OE programme at Koeberg is adequate for identifying, grading, and disseminating internal and external OE and research findings;
- Koeberg has a mechanism for providing effective feedback on OE for ageing management to benefit from both internal and external operating experiences;
- a robust corrective action programme exists. Corrective actions were prioritised, scheduled, and – in general – effectively implemented. Effectiveness reviews were performed, where appropriate;
- sufficient computer hardware and software tools exist to collect, store, retrieve, and document operating experience at Koeberg;
- Koeberg has sufficient sources of OE and had sufficiently incorporated these into the organisation;
- Koeberg conducts investigations of potential trends, and quarterly deviation trend reports are produced and reported to safety committees;
- safety-related events (internal OE) are adequately identified, captured, assessed, and incorporated into organisational learning processes; and
- nuclear and radiological safety performance are adequate as determined by the recording, analysis, and trending of internal operating experience.

9.8.1 Corrective Action Programme

The corrective action process (CAP) is the cornerstone of the performance improvement programme. As such, it will be described in more detail in this section. The corrective action programme at Koeberg is robust and meets national and international requirements. This robustness was found to be a strength in the PSR. The corrective action process is managed in accordance with KAA-688 (*The Corrective Action Process*) [146]. The programme objectives are:

- to describe the process and responsibilities for identifying, reporting, investigating, and trending occurrences, problems, events, conditions, and near misses, as well as ageing-related degradations;
- to ensure that operating experience information is duly recognised, screened, classified, investigated, distributed, and tracked to inform actions to improve nuclear and radiological safety, conventional safety, health, and the environment in order to prevent events from recurring and to ensure continuous improvement; and
- to establish uniform practices for reporting, recording, classifying, investigating, and closing out occurrences, problems, events, conditions, and near misses.

The identification and reporting of occurrences, incidences, conditions, events, or near misses, including ageing-related degradations, are each person's responsibility at the facility. The process administrators are responsible for capturing the relevant coding for their department's items in DevonWay (the software system for capturing, tracking, and managing issues). The CAP process provides a platform for integrated management of all plant events.

9.8.2 Sources of Operating Experience

The process for identifying, grading, and disseminating internal and external OE and research findings is adequate against national and international standards. Processing of internal and external OE is undertaken according to document KAD-025 (*Processing of Operating Experience*) [155].

To demonstrate organisational learning due to the incorporation of OE into processes and procedures, Koeberg has implemented safety improvements from benchmarking with EDF and other industry operators. Eskom is committed to continuously improving the safety of the plant and has adopted the IAEA approach to performing periodic safety reviews (PSRs). Eskom is a member of WANO and INPO and, as such, benefits from having access to a pool of industry OE and peer reviews geared to ensuring a high level of safety performance by evaluating strengths and shortfalls from excellence. Being a member of WANO and INPO Atlanta Centre, Koeberg receives OE such as significant operating event reports (SOERs) and industry event reports (IERs).

The OE is managed and distributed within the organisation in accordance with the CAP process, KAA-688 (*Corrective Action Process*) [146]. Koeberg is a member of EPRI and has a contract with the institute to provide research work on various topics as requested by Koeberg. An example of this is the guidance provided by EPRI to perform an interim seismic evaluation. Koeberg has a contractual agreement with EDF to share various OE such as EDF Affaire Parcs and safety events. The Koeberg governance document 331-23 (*Processing of Industry Operating Experience in Nuclear Engineering*) [106] describes the process, roles, and responsibilities of how the EDF OE is managed within the organisation. Koeberg has an integrated team ("KIT") that is responsible for receiving all OE from EDF and screening it for applicability.

Improvements to the safety of Koeberg include safety improvements resulting from the external events safety reassessment (EE-SRA) based on the Fukushima accident in 2011, CP-1 alignment projects based on EDF findings, adoption of the ageing management matrix following SRA-II based on EDF findings, and others.

9.8.3 Ageing Management Programmes

Regarding the use of OE in ageing management[1], Koeberg considers OE from a wide range of sources. It also considers research findings extensively from EPRI, EDF, and ad hoc from local universities. These sources, together with nuclear industry feedback, provide valuable input to plant safety and ageing management for SSCs throughout the life cycle of the plant.

OE is incorporated into plant programmes, especially the use of significant industry experience. Document 240-149139512 (*Ageing Management Requirements for KNPS*) [52] contains requirements for OE in ageing management, including an annual review of OE and a detailed comparison of Koeberg AMPs and IGALL during the 10-yearly PSR.

The corrective action and OE processes have been integrated into the AMP. “Corrective actions” and “OE feedback and feedback of research and development results” are two of the adopted nine generic attributes of an effective AMP, thus ensuring that ageing anomalies are timeously corrected and that lessons learnt from OE are considered and incorporated.

As per document 331-148 (*Programme Engineer’s Guide*) [97], condition reports are generated when programme results fail to meet acceptance criteria and on detection of unexpected significant ageing degradation. Evaluations are performed according to the requirements of KAA-688 (*Corrective Action Process*) [146], and appropriate actions are taken to enhance preventive actions, identify trends, for monitoring, and inspections, as necessary.

The approach to the identification and review of ageing-related OE is described in ageing management procedure 331-275 (*Process for the Development and Control of Ageing Management at Koeberg Operating Unit*) [107]. Component failure reports from CAP are reviewed periodically to monitor for ageing mechanisms, and the results are used to modify the requirements of the existing ageing management programmes.

Through the evaluation of age-related corrective actions and relevant OE, ageing management programme requirements are modified, and in some cases, new ageing management programmes are developed. A specific event trend code has been created in DevonWay for ageing. This trend code is assigned to ageing-related failures or occurrences.

In summary, there are processes and procedures in place for management of the OE programme and delivery of performance improvement as determined during the PSR safety performance and

[1] Also refer to § 9.5.

operating experience review. The organisational learning processes are well embedded and used effectively to drive performance improvements for ageing management of the plant.

9.9 Nuclear Safety Culture

The regulatory document, RD-0034 (*Quality and Safety Management Requirements for Nuclear Installations*) [290], defines nuclear safety culture (NSC) as the characteristics and attitudes of organisations and individuals that ensure that nuclear safety issues, as an overriding priority, receive the attention warranted by their significance. NSC at the Koeberg NOU is considered acceptable and essential for continued safe operation into LTO.

The section draws on the findings from multiple safety factors during the third PSR and the global assessment to demonstrate that the nuclear operating unit has a healthy NSC as defined in INPO 12-012 (*Traits of a Healthy Nuclear Safety Culture*) [276] to support LTO. The nuclear management policies and the higher-level procedures to support the safety management system will be discussed first, followed by monitoring and oversight of NSC at the NOU.

A healthy NSC enables good human performance and plant safety performance. The findings related to NSC found during the global assessment are provided in § 9.1. Key plant safety performance indicators, including human performance indicators, were reviewed during the third PSR and are discussed in § 9.9.3. Finally, conclusions on the state of the NSC at the nuclear operating unit are drawn.

9.9.1 The Nuclear Management Policy and Safety Management System

The safety management system of the NOU complies with RD-0034 (*Quality and Safety Management Requirements for Nuclear Installations*) [290] and provides the framework for promoting, establishing, and maintaining a healthy NSC at the NOU for the full LTO duration. International Nuclear Safety Advisory Group (INSAG) 13 (*Management of Operational Safety in Nuclear Power Plants*) [247] defines the safety management system as those arrangements for the management of safety to promote a healthy safety culture and achieve good safety performance. The key higher-level documents of the NOU supporting the nuclear management policy and some of the processes for the safety management system are listed below.

- 32-83 (*Nuclear Management Policy*) [90] outlines Eskom management's commitment to promoting a healthy nuclear safety culture by developing and reinforcing good safety attitudes and behaviours in individuals and teams, with nuclear safety and occupational safety being the overriding priority.
- 238-8 (*Nuclear Safety and Quality Management Manual*) [28] integrates all the safety and quality requirements for the nuclear operating unit.

- 238-28 (*Nuclear Safety Management Programme*) [9] provides the key elements of the nuclear safety and human performance system and the responsibilities of leaders and staff to foster a healthy nuclear safety culture.
- KAA-850 (*Koeberg Nuclear Power Station Safety Culture Enhancement Programme*) [151] describes the SCEP development and implementation for Koeberg. This was specifically developed to comply with RD-0034 (*Quality and Safety Management Requirements for Nuclear Installations*) [290].
- 240-108035478 (*Eskom Nuclear Objectives*) [32] provides expectations for nuclear safety, technical performance, organisational effectiveness, and the participation and benchmarking within the nuclear industry. It also describes the operational health dashboard (indicators), which is used to monitor performance.
- In 36-1518 (*Nuclear Safety Oversight in Eskom*) [124], in line with INPO 17-004 (*Principles for Excellence in Corporate Performance*) [277], independent internal and external oversight bodies provide senior managers with an independent view of plant performance nuclear safety. Examples of the oversight bodies are the Nuclear Safety Assurance Department, Nuclear Oversight Department, Nuclear Safety Review Committee (NSRC), World Association of Nuclear Operators (WANO), and Nuclear Safety Review Board (NSRB).

The PSR safety factor on organisation and management systems review [69] provides a detailed assessment of the features of a safety management system, such as safety policy and safety requirements, planning and control, implementation, audits, and review and feedback. The assessment confirmed that safety management system requirements are largely developed, implemented, maintained, reviewed, and monitored. These governing procedures were assessed during the PSR safety performance review, and their implementation compliance is acceptable.

The nuclear operating unit has adopted the 10 'Traits of a Healthy Nuclear Safety Culture' (INPO 12-012) [276], which are at the core of the NSC framework and are shown in Table 9-8.

Table 9-8: *Traits of a Healthy Nuclear Safety Culture*

Traits of a Healthy Nuclear Safety Culture (INPO 12-012)	
Individual commitments to safety	Personal accountability (PA) Questioning attitude (QA) Effective safety communication (CO)
Management commitment to safety	Respectful work environment (WE) Leadership safety values and actions (LA) Decision-making (DM)
Management systems	Continuous learning (CL) Problem identification and resolution (PI) Environment for raising concerns (RC) Work processes (WC)

Several platforms communicate nuclear safety to ensure open, two-way communication on the nuclear management policy, nuclear safety culture awareness, and human performance. Platforms include plant induction training for all staff (including contract staff), nuclear safety awareness sessions done in accordance with the safety culture plan (usually annually), self-study sessions for technical leaders, and monthly (and, in some cases, weekly) employee and senior management engagement sessions. Document 240-131691121 (*Internal and External Communications Procedure*) [42] provides some of the communication channels utilised to communicate the nuclear management safety policy.

The PSR human factors review [73] confirmed that Koeberg complies with the requirements to integrate NSC principles into training programmes for specific plant activities and provide training on the traits of a healthy nuclear safety culture to all staff in accordance with IAEA NS-G-2.8 (*Recruitment, Qualification, and Training of Personnel for Nuclear Power Plants*) [251]. It also confirmed that procedures, for example, KAA-865 (*Human Performance Programme*) [153] and KSA-122 (*Human Performance Tools*) [181], have been implemented to evaluate and manage human performance and human error effectively. This inculcated awareness of the importance of a healthy NSC.

In conclusion, the nuclear management policy and safety management system processes and procedures of the NOU are based on regulatory and international standards. They are well communicated and serve as a robust framework to maintain a healthy NSC during LTO.

9.9.2 Monitoring and Oversight of the NSC at the Nuclear Operating Unit

Monitoring and oversight are essential elements of an effective safety management system. They allow for the early recognition of potential negative trends so that corrective actions can be taken. Monitoring the safety management system (and NSC programme) is mandated by the Eskom nuclear safety management policy described in § 9.9.1. The NSC programme at the NOU is sufficiently monitored, and oversight is provided to identify vulnerabilities and opportunities for improvement. As documented in the KAA-850 (*Koeberg Nuclear Power Station Safety Culture Enhancement Programme*) [151] procedure, the NSC programme is regularly evaluated through self-assessments, quality assurance audits, assessments by the Nuclear Safety Assurance Department, and independent surveys conducted by external organisations. One of them is Inavit IQ, an organisation conducting nuclear culture surveys.

Eskom has various monitoring and oversight committees at all levels within the organisation. Additionally, there are independent, external oversight bodies. The highest-level independent external oversight committee is the Nuclear Safety Review Board, reporting directly to the chief nuclear officer, and the outcomes are shared with the Eskom Board Sub-committee on Social, Ethics, and Sustainability, while the highest-level internal oversight committee is the Nuclear Management Committee, chaired by the Eskom group chief executive. The Nuclear Safety Review Board consists of a team of experienced nuclear professionals, typically at the level of the power station manager, which reviews specific areas of focus, including, but not limited to, operating, maintenance, and engineering.

This oversight structure cascades downwards into various supporting committees, functional groups, and departments. At senior management and executive level, plant safety performance is presented through an operational health dashboard, which includes benchmark indicators from the World Association of Nuclear Operators (WANO). At an operational level, there are several monitoring and oversight committees, such as the Plant Health Committee focusing on plant reliability, the Koeberg Operations Review Committee (KORC) focusing on plant safety and operability, and the station Corrective Action Review Forum and Human Performance Oversight Committee focusing on their specific areas of concern.

Nuclear safety culture surveys are performed three-yearly, utilising INPO 12-012 (*Traits of a Healthy Nuclear Safety Culture*) [276]. The surveys were conducted in 2014, 2016, and 2019 and submitted to the NNR. The 10 traits of a healthy NSC, each with its attributes and behaviours, are clustered into three broad categories (see Table 9-8). Comparing the results of the NSC surveys revealed that the score for all traits had improved from 2014 to 2019 [59]. Recommendations stemming from the 2019 NSC survey were consolidated into three improvement actions and have since been implemented. The improvement actions related to communication and engagement strategies on nuclear safety across all levels of the organisation, visible and consistent rewards and recognition for good behaviours, and organisational leadership team development. These are assessed for effectiveness during the annual NSC self-assessment.

A quality assurance audit on nuclear safety culture completed in May 2020 is worthy of note, as it was rated “not met” [126]. There were 15 non-conformances, none of which had a significant impact on process objectives, and four non-conformances were rated as having a material impact on process objectives. These non-conformances are being addressed via the quality assurance process. They are being investigated, and corrective actions will subsequently be prescribed and implemented, followed by verification of effective action and closure by the Quality Assurance Department.

In summary, the NSC programme is well established at Koeberg. As discussed above, trends in the traits that constitute a healthy nuclear safety culture are sufficiently monitored, and oversight is appropriately provided for the early detection of potential negative trends to recognise opportunities for improvement and arrest adverse trends.

9.9.3 Nuclear Objectives and Associated Plant Safety Performance and Human Performance Indicators

In line with RG-0007 (*Regulatory Guide – Management of Safety*) [291], safety goals and objectives must be defined to demonstrate commitment to quality, safety performance, and a healthy nuclear safety culture. Clear nuclear safety objectives aligned with international practices have been established for Koeberg and are contained in procedure 240-108035478 (*Eskom Nuclear Objectives*) [32]. It lists several indicators that can be used to monitor improvement in safety performance to achieve Eskom's safety objectives. Examples of the indicators being monitored (from which it can be inferred that the NOU safety objectives are being met) are peak public risk, core damage frequency, availability of safety systems, unplanned automatic grid separations, etc.

These trends were reviewed in the PSR safety performance review from 2009 to 2019 [73]. The review concluded that the safety performance trends were stable with isolated incidents, appropriately identified, investigated, and corrected.

The PSR safety performance [73] and human factor [70] reviews included human performance indicators. While several human performance trends were positive, some indicated a negative trend. Continuous improvement efforts will address the negative trends; however, the overall assessment was acceptable for human performance.

In summary, as discussed above, clear nuclear safety objectives aligned with international practices have been established. A set of plant performance and human performance indicators are being monitored. Based on the trends observed, safety and human performance at the nuclear operating unit are acceptable, and negative trends are appropriately recognised and addressed.

9.9.4 PSR Global Assessment Outcomes Related to NSC

Multiple factors were considered in the review of the NSC. The PSR global assessment [115] provided a critical look at where continuous improvement was needed to address indications of potentially inadequate NSC. It considered 113 deviations across 14 safety factors. The tasks included causal analysis, defence-in-depth analysis, and interface analysis based on the IAEA fundamental safety principles. It found eight cross-functional global issues, of which three global issues were considered most relevant to NSC, and these are discussed below.

- Issue related to insufficient management oversight, inadequate use of, and adherence to, procedures, and inadequate processes

  This global issue was graded "low". There were 17 deviations directly linked to this global issue, approximately 15% of all deviations across all PSR safety factors. The deviation gradings ranged from "low" to "drop" and, consequently, did not significantly affect plant safety. Improvement actions to address the issue included training for managers, including leadership skills development, and capacitating managers to make decisions.

- Issue related to inconsistent demonstration of some NSC traits of a healthy NSC

  The global issue highlighted seven deviations related to some of the 10 traits of a healthy NSC as defined in INPO 12-012 (*Traits of a Healthy Nuclear Safety Culture*) [276]. This global issue was graded "low". The seven deviations directly linked to this global issue represented approximately 6% of all deviations across all safety factors. Among the seven deviations, personal accountability and leadership values and actions were the two traits identified for improvement actions. Improvement actions to address the issue included an assessment of the affected NSC traits to determine specific actions and communication sessions on the findings related to NSC. The deviations were all graded "low" and did not significantly affect plant safety.

- Issue related to the culture of tolerance of long-standing issues

  This global issue was found through the third PSR interface analysis using the fundamental safety principles in the PSR plant design review. The basis for the global issue was the delayed resolution of eight deviations, some of which stemmed from SRA-II, and was linked to Fundamental Safety Principle 3 (*Leadership and Management for Safety*). This global issue was graded "medium". However, the fundamental safety principles were judged to be met, as the leadership and management systems remained effective despite the impact of the deviations, including their cumulative effect (see Appendix B of the global assessment report). Furthermore, progress had been made in implementing some mitigating actions to reduce the impact of these deviations. The improvement actions were well defined, tracked, and monitored.

In summary, the global assessment considered, through various analyses, whether the causes of deviations were linked to weaknesses in the NSC. Only a small proportion of deviations was directly linked to the global issues mentioned above, which are related to weaknesses in NSC. Improvement actions have been prescribed and included in the PSR IIP to address the deviations.

While opportunities for improvement exist, based on the outcome of the PSR as documented in the PSR safety factor on organisation and management systems review [69] and global assessment report [115], as well as the arguments presented above, NSC at the nuclear operating unit is acceptable and appropriately monitored for continued safe operation into LTO.

10.0 Why it is Safe to Continue Operation (Overall Assessment for Additional 20 Years)

This section demonstrates that Koeberg is ready for continued safe operation for an additional 20 years.

- A comprehensive PSR to determine, among others, the extent of the safety of the plant when compared to modern codes and standards and the validity of the current licensing basis was performed. ***Fulfils RG-0027 (5.3.1); R.266 4 (c), (e).***
- The PSR confirmed that the overall safety of the plant was adequate and that the level of safety would be maintained and/or improved with the implementation of the identified safety improvements. It also confirmed that the plant would be suitable for continued operations, provided that the safety improvements were to be implemented. ***Fulfils RG-0027 (5.3.1); R.266 4 (c), (e).***
- In the past 39 years, the principal safety criteria and requirements have been respected. The criteria will continue to be respected, since plant operations will remain largely unchanged for the intended period of LTO. The facility complies with the NNR-established risk limits, and the LTO assessments have demonstrated that the facility has adequate structures and processes to ensure continued compliance with these limits in the LTO period. Current regulations require that any changes to the operations of the plant that affect the principal safety criteria be approved by the NNR. ***Fulfils RG-0027 (5.3.1); R.266 4 (a), (c).***
- It has been demonstrated that the design of the plant is adequate, and life-limiting SSCs (note: no structures requiring replacement were identified) important to safety have been identified for replacement. No design-related concerns were found that precluded the plant from safe LTO. ***Fulfils RG-0027 (7.1); R.266 4 (e).***
- It has been demonstrated that the plant, as far as reasonably practicable, has robust defence in depth and enough independencies of the various levels of defence, which are key in managing accidents and, thus, ensuring no undue risk to the public and the environment. ***Fulfils RG-0027 (5.3.1); R.266 4 (a).***
- Additionally, the safety of the plant has been enhanced through the application of the concept of defence in depth in various plant processes (such as multiple reviews, oversight, etc.). ***Fulfils RG-0027 (5.3.1); R.266 4 (a).***
- The Koeberg ageing management philosophies and practices have been demonstrated to be aligned with national and international codes and standards. Deviations with low safety significance related to requirements for entry into LTO were identified and assigned appropriate corrective actions. All ageing management actions resulting from ageing management assessments have been prioritised into actions required to be implemented before the end of

40 years of operation, which are currently in progress, with the remainder to be implemented in the LTO period. ***Fulfils RG-0027 (7.1); R.266 4 (b), (e)*.**

- It has been demonstrated that operational programmes related to safety, health, and the environment, such as emergency planning and response, environmental management, radioactive waste management, and others, remain adequate and effective to support safe LTO and that LTO has an insignificant impact on these programmes due to facility operational regimes remaining unchanged. ***Fulfils RG-0027 (7.1); R.266 4 (a), (c)*.**
- Although the site safety studies are being revised using the latest methodologies, the risks associated with the changes in external hazards have been identified, quantified, and where necessary mitigated. The interim evaluation of the robustness of the plant against seismic hazards (using the ESEP methodology) concluded that, with the implementation of identified safety improvements, the plant could withstand the impact of such a hazard. Additionally, the interim tsunami probability analysis indicated that the risk associated with the tsunami hazard is low. Further detailed seismic and tsunami analyses are in progress and will be completed prior to LTO. ***Fulfils RG-0027 (5.3.1 and 7.1); R.266 4 (b), (c)*.**
- Regarding emergency preparedness and response, a review of the technical basis for the emergency plan demonstrated that the emergency plan was adequate for safe LTO. ***Fulfils RG-0027 (5.3.1); R.266 4 (b), (c)*.**
- There is a mature, comprehensive integrated management system in line with international practices, which ensures that risks will be adequately managed in the LTO period. ***Fulfils G-0027 (7.1); R.266 4 (e).***
- It has been demonstrated that there are processes to ensure that sufficient resources (human, financial, and equipment) are available throughout the life of the plant. ***Fulfils RG-0027 (5.3.1); R.266 4 (d)*.**
- Comprehensive training and knowledge management programmes are in place to manage the skills and expertise in the LTO period. The implementation of the KM programme is in progress and is due to be completed prior to entry into LTO. ***Fulfils RG-0027 (5.3.1); R.266 4 (d)*.**
- An LTO integrated implementation plan, detailing all the safety improvements for safe LTO, has been developed. ***Fulfils RG-0027 (7.1); R.266 4 (e).***

It is safe to continue operations since it has been demonstrated that nuclear safety at the facility will be maintained in accordance with regulatory requirements and international good practices for the intended period of LTO, provided that there is timely implementation of the safety improvements contained in the LTO IIP.

11.0 Safety Analysis Report

This section discusses the proposed changes to the current SAR as directed and required by the NNR guidance documents.

The SAR is recognised as a "living" document and comprehensively reflects the plant design of Koeberg. The SAR is kept current and aligned with the plant design by continually updating changes resulting from plant modifications, licence-influencing changes, and any process changes affecting the licence, using the SAR update procedure 240-119744497 (*Control of the Safety Analysis Report*) [37].

The PSR plant design review assessed the SAR as a design basis document for content and comprehensiveness. No deviations were identified; however, there was a deviation raised regarding codes and standards (as discussed in § 9.4), which might result in SAR updates.

The NNR issued RG-0019 (Guidance on the Safety Assessments of Nuclear Facilities) [293] to guide users in complying with nuclear safety criteria and licensing requirements. RG-0019 provides specific requirements relating to the SAR. The current Koeberg SAR does not meet the format and content of RG-0019, Appendix 4. Eskom indicated, in correspondence letter K-28083-E, that the Koeberg SAR would meet all the content according to RG-0019, but not the format in Appendix 4. The NNR accepted Eskom's proposal in correspondence letter k28083N on condition that Eskom maintain the reconciliation RG-0019 requirements.

In section 5.3.1 and section 7.9, RG-0027 states that an updated SAR should be included in the documents supporting LTO. The updated SAR should also include documents of the revalidation of the time-limited ageing analyses (TLAAs) for the period of long-term operation.

Additionally, the SAR for the current life of the plant defines the design service life of the nuclear steam supply system (NSSS) as 40 years. Therefore, the SAR must be revised with the analysis for a 60-year service life of the NSSS.

At the submission of the safety case, some of the documentation required for the SAR updates was not available, since the activities related to these are currently in progress, and scheduled to be completed prior to entry into LTO. In accordance with the approval obtained from the NNR, the updated SAR with markups for the outstanding activities will be submitted. Therefore, a marked-up SAR is submitted to support the safety case. Appendix B provides details regarding the proposed changes to the SAR.

Regarding the SAR changes related to the SGR and SALTO project safety analyses, Table 11-1 is provided, indicating the Koeberg internal approval schedule for the analyses.

Table 11-1: *SGR and SALTO TLAA SAR Changes*

SAR UPDATE PROGRESS									
	Total Number of TLAAs Reports	**Total Number of Changed Chapters**	**Change Notices (CN) Generated**	**Change Notices not yet Generated**	**CNs Under Safety Documentation Review Group (SDRG) Review**	**SDRG Approved CNs)**	**Submitted to NNR**	**Projected submission to NNR (TLAA Reports Numbers)**	
							4 July 2023	30 November 2023	31 January 2024
SGR changes – completed	N/A	101	101	0	0	101	All SGR SAR Changes submitted to NNR by the end of 2022 and approved by NNR by August 2023.		
TLAA changes	29	6	1	2	0	1	5	13	11

12.0 Adopted Long-Term Operation Programme

This section describes the elements of the long-term operation (LTO) programme to be adopted for future operations during LTO. The LTO programme must meet the requirements in section 7.4 of the NNR's RG-0027 [294] and the NIL-01 condition 16.2 and demonstrate how it will be maintained for future operations.

RG-0027 requires that the comprehensive programme for LTO address the following:

- Preconditions (including the current licensing basis, safety upgrading and verification, and operational programmes)
- Setting the scope for all SSCs important for nuclear safety
- Categorisation of SSCs with regard to degradation and ageing processes (commodity grouping)
- Revalidation of safety analyses made based on time-limited assumptions
- Review of ageing management programmes in accordance with current NNR and international safety standards and operating practices
- The implementation programme for LTO

The preconditions were verified during the PSR performed in support of LTO, and no deviations were raised relating to the preconditions. Initial scope setting, commodity grouping, revalidation of TLAAs, and review of AMPs were performed in the SALTO ageing management assessments.

However, these activities will be required throughout the extended period of operation when plant changes are made, and as discussed in § 9.5, the framework exists to ensure that these are performed as required.

Most of the identified TLAAs were confirmed valid for the entire period of LTO. The EQ TLAA could not be revalidated for the entire period, and for all the equipment. Therefore, a replacement plan was developed for the equipment throughout the life of the plant. The replacements were incorporated into the AMP updates. These replacements are detailed in Appendix A of this report.

For TLAAs where the revalidation is pending, ageing management actions had been identified to ensure continued safe operation if they were to not be valid for the additional 20 years. These actions are detailed in § 9.5 of the report.

The AMPs were reviewed based on available information. However, the improvement of these was largely based on operating experience, and therefore, continuous review will be performed during the period of LTO.

To ensure that the elements of the LTO programme mentioned above are adequately addressed throughout the life of the plant, the following is applicable:

- Based on adopting the ageing management approach described in RG-0027, Koeberg developed an AM standard to ensure that all required AM activities were implemented according to the regulatory guidance document.
- Document 36-197 (*Koeberg Licensing Basis Manual*) [125] was updated to reflect the ageing management standard, as the KLBM is utilised to demonstrate the processes applicable to the fulfilment of NIL-01 Variation 19. Document 36-197 was revised recently, is awaiting NNR approval, and is tracked under CR110967-001CA.
- It was demonstrated in § 9.5 that AM activities required for LTO have a suite of procedures that govern the performance of these activities to ensure that they were performed systematically.
- The additional AM actions identified (such as replacements, inspections, and maintenance) in the assessments that had to be performed during the LTO period were incorporated into the plant AMP.

13.0 Long-Term Operation-Related Documents

According to RG-0027 [294], the assumptions, activities, evaluations, assessments, and results of the plant programme for LTO should be documented by the operating organisation according to current national or international safety standards.

Regarding ageing management, the documents should include the following to demonstrate that ageing effects will be managed throughout the planned period of LTO:

- A description of safety-related programmes and documents relevant to ageing management throughout the planned period of LTO
- A list of activities for the improvement or development of safety-related programmes and documents relevant to ageing management throughout LTO as well as information on the implementation of new ageing management programmes
- The documents should include an update of the safety analysis report and other documents required by the licensing process reflecting the assumptions, activities, and results of the plant programme for LTO. The update to the safety analysis report should also include documents concerning the revalidation of the time-limited ageing analyses for the period of LTO.

The following Eskom documents related to LTO respond to the requirements above:

- 331-608 (*KNPS 3rd Periodic Safety Review Global Assessment Report and* Integrated *Implementation Plan Report*) [115], which considers all the PSR deviations and demonstrates the suitability for continued operations based on an assessment guided by KGA-029 (*Safety Justification Preparation*) [170].
- 240-156945472 (*SALTO Ageing Management Assessment Report (Interim)*) [57], which documents the safety assessment of the ageing management aspects performed at Koeberg to

achieve the regulatory ageing management requirements and provide assurance for safe LTO. The SALTO final report will be issued before LTO.

- *Koeberg Safety Analysis Report (SAR)* [178], which is periodically revised to include changes that result from modifications to the plant, operational experience feedback, the correction of errata, and the results of new safety analyses.
- 36-197 (*Koeberg Licensing Basis Manual*) [125], which defines the licensing basis and gives the key mandatory nuclear safety principles and documents that must be complied with to control and demonstrate the safe operation of Koeberg. The KLBM is structured to reflect the main groupings of requirements and processes adopted to provide assurance regarding the safe operation and decommissioning of the Koeberg Nuclear Power Station according to a licensing basis.
- 240-160692496 (*Long-Term Operation Programme Management Manual*) [62], which defines the programme organisation, roles and responsibilities, licensing interfaces and schedule, resources, and deliverables according to RG-0027 [294].
- K08016VAR (*Koeberg Plant Life Extension*) [133]
- 240-150483693 (*Ageing Management Programmes List*) [53]
- 240-106374672 (*Koeberg Pre-SALTO Self-Assessment Report*) [31]
- 240-100984199 (*Koeberg Long-Term Operating (LTO) Methodology*) [29], which provides the methodology that Eskom uses for performing the nuclear safety assessment and associated safety demonstration for Koeberg to continue operation until at least 2045.
- 240-132364298 (*Initial List of Time-Limited Ageing Analyses for Koeberg Nuclear Power Station*) [43], which describes the initial TLAAs applied to ageing management evaluation (AME) activities for the plant.
- 240-125122792 (*Koeberg Safety Aspects of Long-Term Operation (SALTO) Ageing Management Evaluation Process and Revalidation of the Time-Limited Ageing Analyses*) [39], which describes the process for assessing the SSCs identified for ageing management evaluation to demonstrate that ageing degradations and effects will be effectively managed and monitored for the planned period of LTO for the in-scope SSCs and to verify the adequacy of programmes and processes used to manage and mitigate these ageing effects. The document, furthermore, provides the process for identification and revalidation of the time-limited ageing analyses (TLAAs).
- 240-125839632 (*Koeberg Long-Term Operating (LTO) Scoping Methodology*) [40], which provides a systematic scoping process to identify SSCs subjected to the SALTO ageing management evaluation and outlines the process followed during the scoping activities to meet the scoping regulatory requirements in accordance with RG-0027.

- 240-149139512 (*Ageing Management Requirements for Koeberg Nuclear Power Station*) [52], which provides overall requirements for the ageing management of safety-related equipment and indicates the links with related physical ageing management processes at the station for the life of the plant, including LTO. It covers all stages of the equipment life of the plant, that is, design, construction, manufacturing, commissioning, operating, LTO, suspended operation, and decommissioning.
- 331-607 (*Periodic Safety Report Final*) [114], which provides a summary of the outcomes from the review, including a list of findings indicating areas where current standards and practices are not achieved and a list of areas where current safety standards and practices are exceeded (that is, plant strengths), a summary of the outcomes from the global assessment, and an integrated implementation plan of proposed safety improvements, including their safety significance and prioritisation.
- LTO IIP: Appendix A has been provided, indicating the schedule of safety improvements.

14.0 Long-Term Operation Integrated Implementation Plan

The safety improvements resulting from the LTO assessments are categorised into two groups, namely, LTO integrated preparation plan, which are safety improvements required prior to entry into LTO (mainly supporting the justification arguments), and LTO implementation plan, which are safety improvements required to ensure safe LTO during the LTO period. This categorisation is in accordance with the NNR approved safety case structure and content document. All the safety improvements are contained in the LTO IIP. Therefore, this section aims to consolidate and document all activities for the LTO IIP.

§ 14.1 (*LTO integrated preparation plan*) provides all the safety improvements that must be performed before entering LTO. These actions are discussed in the justification presented in § 9.0 and support the conclusions of the safety case.

§ 14.2 (*LTO implementation plan*) provides all the safety improvements required in the LTO period, that is, after the licence extension has been granted and Koeberg has entered the LTO period.

The information presented in § 14.1 and § 14.2 has been consolidated from the following:

- Interim SALTO ageing assessment and SALTO peer review outcomes (knowledge management commitments)
- Koeberg's third periodic safety review global assessment report and integrated implementation plan
- Site safety report and interim seismic hazard assessment for LTO
- Outcomes of the assessment of other safety-related programme assessments for LTO
- Plant life extension feasibility study

14.1 LTO Integrated Preparation Plan

331-608 (*KNPS 3rd Periodic Safety Review Global Assessment Report and Integrated Implementation Plan Report*) [115] lists the outcomes of the ageing management assessments. It includes actions such as TLAA revalidations, implementation of AMPs, modifications, and replacements to remove life-limiting plant features. These are safety improvements required to meet the ageing management requirements in RG-0027 [294]. Included in this list is the PSR IIP safety improvements identified as directly linked to the 40-year end-of-life and safety improvements as a result of the interim seismic evaluation.

Table A.1-2 lists the DSSR and interim seismic hazard analysis-related activities.

331-608 (*KNPS 3rd Periodic Safety Review Global Assessment Report and Integrated Implementation Plan Report*) [115] also lists the PSR IIP safety improvements not related to ageing, but required prior to entry into LTO.

Appendix B lists the SAR updates for other safety-related programmes.

Table A.1-3 lists the TLAAs reanalysed and currently being processed for NNR submission.

14.2 LTO Implementation Plan

Table A.2-1 lists the ageing management safety improvements that are required to support safe LTO and can be completed during LTO. The AMPs will be implemented in accordance with current work management processes KAA-721 (*Online Work Management Process*) [147] and KAA-829 (*Development of an Outage Plan*) [150].

Table A.2-2 lists the PSR IIP safety improvements to be completed during LTO.

The execution and monitoring of these activities will be in accordance with the organisational mandates as prescribed in 240-64602879 (*Nuclear Operating Unit Organisational Structures*) [81].

In addition, the implementation of the ageing management programmes is monitored and overseen in accordance with 240-139089079 (*Programmes Oversight Committee – Terms of Reference*) [45]. Document 331-148 (*Programme Engineer’s Guide*) [97] provides the requirements for the process for monitoring, developing, maintaining, optimising, and performing programme oversight activities on the ageing management programmes. The strategic Plant Health Committee also provides oversight of all plant-health-related activities in accordance with KAA-826 (*Plant Health Committee (PHC) Constitution*) [149].

Table A.1-2 lists the DSSR and interim seismic hazard analysis-related activities.

15.0 Conclusions

Eskom has conducted comprehensive assessments and studies to assure the Regulator of Eskom's commitment to meeting the national LTO requirements, the national safety criteria, national and international safety standards and codes, for the period of LTO. The safety case provides documented arguments and evidence of Koeberg's suitability for LTO. Elements of the safety case, as argued in this document, include:

- robustness of the plant design for the intended period of operations;
- effectiveness of the ageing management of SSCs important to safety;
- implementation of ageing management programmes;
- implementation of the DiD in the design and operation of the nuclear installation;
- adequacy of emergency planning, radiation protection, and waste management programmes;
- provisions to ensure continued compliance with occupational, public, and environmental safety requirements;
- adequacy of organisational arrangements for safe LTO, including knowledge management, human, and financial resource arrangements, with an emphasis on ageing management; and
- adequate use of OE for continuous improvement.

Additionally, the safety case summarises why it is safe to continue operations. Eskom has identified safety improvements required to address shortfalls to meet regulatory requirements and to ensure the continued safety of Koeberg. These safety improvements are categorised into those required to be implemented before entering the period of LTO to ensure the validity of the safety case and those that will be implemented during the LTO period, as part of the LTO programme.

Eskom has made adequate progress on the implementation of the LTO IPP actions. It has been demonstrated that nuclear safety at the facility will be maintained in accordance with regulatory requirements and international good practices for the intended period of LTO. Therefore, through the safety assessments that have been conducted as well as the implementation of the safety improvements, it can be concluded that Eskom meets the LTO requirements as stipulated in R.266 [240].

As guided by RG-0027 [294] and document 240-157754316 (*Structure and Content of the Safety Case*) [58] accepted by the NNR, Eskom hereby submits the safety case to the NNR for consideration.

16.0 Acceptance

This document has been seen and accepted by:

Name	Designation
Danie Moller	Senior Manager Nuclear Project Management
Frikkie Ellis	Senior Manager Nuclear Support
Marc Maree	Senior Manager Licensing (Acting)
Vernon Paul	Plant Manager (Acting)

17.0 Revisions

Date	Rev.	Compiler	Remarks
July 2022	1	Bravance Mashele	This is the first compilation. The document has been compiled to justify the continued safe operation of Koeberg in support of the licence application for long-term operation.
Jan 2023	1a	Bravance Mashele	Minor editorial changes as part of regulatory review process.
Feb 2023	2	Bravance Mashele	The document updated to address NNR comments in letter, k28741N.
Oct 2023	3	Bravance Mashele	The document updated to address NNR comments in letter, k29159N, editorial errors corrected, and provision of progress status for LTO preparation activities.

18.0 Development Team

The following people were involved in the development of this document:

- Fagrie Hendricks : Safety Case Lead Engineer
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- Deon Jeannes : Environmental SME
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- Anita Kilian : Siting SME

- Ditsietsi Malale : Manager Engineering Section
- Bradley Oaker : Engineering Consultant

19.0 Acknowledgements

See §18.0.

Appendix A
LTO Integrated Implementation Plan

A.1 LTO Integrated Preparation Plan (Activities Required for LTO)

Table A.1-1: *Ageing Management LTO Preparation Activities*

Activity Title		Activity Description	Comment	Completion Date
Organisation and Management				
SAR Updates for LTO		Refer to Appendix B for a list of SAR updates for LTO.	Refer to Appendix B.	Refer to Appendix B
Compile and submit the final SALTO Ageing Assessment Report (Final)		SALTO project to submit final ageing management report for LTO to the NNR.	An interim report was accepted by NNR. Final report will be submitted on completion of SALTO activities. Final report is on track.	Dec 23
Update the design classification database with in-scope SSCs that are subject to ageing management requirements in accordance with RG-0027		Update the classification database. Institute a governance procedure for the database of in-scope SSCs.		Complete
Review of EPTB		Reviewing of EPTB to include multi-unit events.		Complete
Mechanical TLAAs				
Environmentally assisted fatigue	Reactor coolant pumps	• Complete reanalysis of TLAA	Screening of TLAA is complete. EAF analysis is in progress.	Jan 24
Environmentally assisted fatigue	Reactor pressure vessel internals	• Complete reanalysis of TLAA	Preparation of TLAA package for submission to NNR in progress.	Complete

Table A.1-1: *Ageing Management LTO Preparation Activities*

Activity Title		Activity Description	Comment	Completion Date
Environmentally assisted fatigue	Pressuriser	• Complete reanalysis of TLAA	Screening of TLAA is complete. EAF analysis is in progress.	Dec 23
Environmentally Assisted Fatigue	Main coolant lines	• Complete reanalysis of TLAA	Screening of TLAA is complete. EAF analysis is in progress.	Jan 24
Environmentally Assisted Fatigue	Auxiliary lines	• Complete reanalysis of TLAA	Screening of TLAA is complete. EAF analysis is in progress.	Jan 24
Environmentally Assisted Fatigue	Control rod drive mechanism	• Complete reanalysis of TLAA	Screening of TLAA is complete. EAF analysis is in progress.	Jan 24
Environmentally Assisted Fatigue	Pressuriser heater sleeves	• Complete reanalysis of TLAA	Preparation of TLAA package for submission to NNR in progress.	Complete
Crack growth analysis of flaws detected in service -AND-Fatigue and thermal ageing analysis of manufacturing flaws and flow tolerance	Pressuriser spray nozzles	• Complete reanalysis of TLAA • Update SAP if required • Complete additional associated TLAA actions if required	Transient analysis and material aspect is complete. Acceptance of Methodology Report and Mechanical Analysis from contractor is in progress.	Jan 24
Reactor pressure vessel internals – thermal ageing and neutron embrittlement	RPV Internals	• Develop TLAA (Fast Fracture Analysis)	Thermal ageing analysis is complete. Fast Fracture analysis is in progress.	Jan 24
Reactor pressure vessel internals – vibration	RPV Internals	• Complete reanalysis of TLAA	Hydraulic load calculations are complete. Dynamic and impact analysis is in progress.	Dec 23

Table A.1-1: *Ageing Management LTO Preparation Activities*

Activity Title		Activity Description	Comment	Completion Date
Mechanical – New Ageing Management Programmes				
Selective leaching	Various	• Implementation of AMP requirements on SAP (creation of service notifications and update of the working procedures and scheduling for execution)		Complete
One-time inspections	Various	• Implementation of AMP requirements on SAP (creation of service notifications and update of the working procedures & scheduling for execution)		Complete
One-time inspections class 1 SB piping	Piping	Implementation of AMP requirements on SAP (creation of service notifications and update of the working procedures & scheduling for execution)		Complete
Primary pressuriser	Pressuriser	• Develop AMP		Complete
Inspection of internal surfaces and internal coatings and linings	Various	• Implementation of AMP requirements on SAP (creation of service notifications and update of the working procedures & scheduling for execution)		Complete

Table A.1-1: *Ageing Management LTO Preparation Activities*

Activity Title		Activity Description	Comment	Completion Date
Heat exchanger programme	Various	• Implementation of AMP requirements on SAP (creation of service notifications and update of the working procedures & scheduling for execution)		Complete
Mechanical - Existing Ageing Management Programme Update				
Preventive maintenance programme - Update the PM programme with the required ageing management activities and implement	Various	• Update existing PM Programme		Complete
RPV Programme	RPV	• Update existing AMP		Complete
Mechanical – One-Time Inspections				
Orifice cavitation erosion inspections on downstream piping according to the SALTO scope as identified through Mechanical AME review (sample)	Orifice **Unit 1** 1LHQ001DI; 1LHQ004DI; 1LHQ005DI; 1LHQ008DI; 1LHQ009DI; 1RCP018DI; 1RCV101DI; 1ASG001DI; 1ASG011DI; 1LHQ002DI **Unit 2** 2REA010TY; 2RRI012DI; 2RRI014DI; 2RRI026DI;	• Perform inspection, evaluate the results, and raise appropriate actions, if required, to address any findings.	Some inspections were completed in outage 125 and 225. The remaining inspections are on track. .	Dec 23 - Unit 1 Outage 226 - Unit 2

Table A.1-1: *Ageing Management LTO Preparation Activities*

Activity Title		**Activity Description**	**Comment**	**Completion Date**
	2RRI033DI; 2RRI044DI; 2RRI045DI; 2RRI053DI; 2RRI055DI; 2RRI101KD; 2RRI131KD; 2RRI133KD; 2RRI135KD; 2RRI141KD; 2RRI303KD; 2RRI341KD; 2RRI351KD; 2RRI361KD; 2DVK001DI; 2PTR001DI; 2PTR007KD **Unit 9** 9TEP031DI; 9TEP032DI; 9LHS001DI; 9LHS002DI; 9LHS004DI; 9LHS008DI; 9LHS009DI; 9LHS005DI			
One-time inspection of cabinet bolting (sample)	Bolting (cabinets)	• Perform inspection, evaluate the results, and raise appropriate actions, if required, to address any findings	Approximately 60% of cabinet bolting inspections are complete. The remaining inspections are on track.	Dec 23 - Unit 1 Outage 226 - Unit 2
One-time inspection of the sampled components of the closed treated water systems	Various	• Perform inspection, evaluate the results, and raise appropriate actions, if required, to address any findings.	All inspections have been reprioritized for online. All inspections are on track.	Dec 23 - Unit 1 Outage 226 - Unit 2

Table A.1-1: *Ageing Management LTO Preparation Activities*

Activity Title		Activity Description	Comment	Completion Date
One-time inspection of the mechanical components sampled from various commodity groups according to the SALTO AME	Various	• Perform inspection, evaluate the results, and raise appropriate actions, if required, to address any findings.	Some inspections were completed in outage 125 and 225. The remaining inspections are on track.	Dec 23 - Unit 1 Outage 226 - Unit 2
Mechanical Component Replacements				
Refurbish SG snubbers on Unit 1	Snubbers for: 1 RCP 001 GV 1 RCP 002 GV 1 RCP 003 GV	• SAP notification raised for refurbishment for the current snubbers	Unit 1 snubbers refurbished in Outage 126.	Complete
Mechanical – Modifications				
Steam generators – Unit 1	1 RCP 001 GV 1 RCP 002 GV 1 RCP 003 GV	• Replace steam generators	Unit 1 steam generators replaced in Outage 126.	Complete
Steam generators – Unit 2	2 RCP 001 GV 2 RCP 002 GV 2 RCP 003 GV	• Replace steam generators	Implementation plans are still on track.	Outage 226

Table A.1-1: *Ageing Management LTO Preparation Activities*

Activity Title		Activity Description	Comment	Completion Date
Electrical – Equipment qualification replacements (TLAA 201)				
Equipment qualification – EQ Components on Unit 1	1 KRT 022 MA 1 KRT 023 MA 1 RCP 001 PO (140/141 MC) 1 RCP 002 PO (240/241 MC) 1 RCP 003 PO (340/341 MC) 1 DVK 100 MP 1 DVK 101 MP	Replacement of the following equipment performed on unit 1: • 1 KRT 022 MA • 1 KRT 023 MA • 1 RCP 001 PO (140/141 MC) • 1 RCP 002 PO (240/241 MC) • 1 RCP 003 PO (340/341 MC) • 1 DVK 100 MP • 1 DVK 101 MP		Complete
Equipment qualification –EQ Components on Unit 2	2 KRT 022 MA 2 KRT 023 MA 2 ETY 201 MP 2 ETY 202 MP 2 RCP 001 PO (140/141 MC) 2 RCP 002 PO (240/241 MC) 2 RCP 003 PO (340/341 MC) 2 VVP 127 VV – EL 1/2 2 VVP 128 VV – EL 1/2 2 VVP 129 VV – EL 1/2	Replacement of the following equipment on unit 2: • 2 KRT 022 MA • 2 KRT 023 MA • 2 ETY 201 MP • 2 ETY 202 MP • 2 RCP 001 PO (140/141 MC) • 2 RCP 002 PO (240/241 MC) • 2 RCP 003 PO (340/341 MC) • 2 VVP 127 VV – EL 1/2 • 2 VVP 128 VV – EL 1/2 • 2 VVP 129 VV – EL1/2 • 2 DVK 100 MP	SAP notifications are raised for execution. Most spares arrived on site. Remaining spares on track for Unit 2, outage 226. .	Outage 226

Table A.1-1: *Ageing Management LTO Preparation Activities*

Activity Title		Activity Description	Comment	Completion Date
	2 DVK 100 MP 2 DVK 102 MP	• 2 DVK 102 MP		
Equipment qualification – requalification of qualified cables	Qualified cables	• Perform the re-qualification of EQ cables for LTO.		Complete
Equipment qualification – EQ Rotork actuators (for MOVs on both units)	Rotork valve actuators	• Perform the overhaul of the qualified Rotork Actuators to renew qualification for LTO	Unit 1 overhaul performed in Outage 126	Complete
			Rotork actuators for unit 2 is on track.	Outage 226
Equipment qualification – MV Cables on RRA Motors on Unit 2	2 RRA 001 MO and 2 RRA 002 MO	• Replace the Medium Voltage Cables connected 2 RRA 001 MO and 2 RRA 002 MO prior to LTO	SAP notifications are raised for outage 226. Purchase request raised.	Outage 226
Equipment qualification – 10RO renewal of the RPN power range detectors and connection plates	1 RPN 010 MA 1 RPN 020 MA 1 RPN 030 MA 1 RPN 040 MA	Renew the RPN power range detectors and connection plates: • 1 RPN 010 MA • 1 RPN 020 MA • 1 RPN 030 MA and • 1 RPN 040 MA	Unit 1 renewal performed in Outage 126.	Complete
	2 RPN 010 MA 2 RPN 020 MA 2 RPN 030 MA 2 RPN 040 MA	Renew the RPN power range detectors and connection plates: • 2 RPN 010 MA • 2 RPN 020 MA • 2 RPN 030 MA and • 2 RPN 040 MA	Spares on site for unit 2 and ready for installation in outage 226	Outage 226

Table A.1-1: *Ageing Management LTO Preparation Activities*

Activity Title		Activity Description	Comment	Completion Date
Electrical - New Ageing Management Programmes				
Environmental condition monitoring programme	Qualified cables and qualified equipment	• Implementation of AMP requirements on SAP (creation of service notifications and update of the working procedures & scheduling for execution)		Complete
Ageing Management Programme for the Electronic Equipment Whiskers and Capacitors	Electronic components	• Implementation of AMP requirements on SAP (creation of service notifications and update of the working procedures & scheduling for execution)		Complete
Ageing Management Programme for Lightning Protection and Grounding Grid at KNPS	Lightning protection systems and grounding systems	• Implementation of AMP requirements on SAP (creation of service notifications and update of the working procedures & scheduling for execution)		Complete
Ageing Management of Switchboards Associated Switchgear Components and their Metal Enclosures	Switchgear and cabinets	• Implementation of AMP requirements on SAP (creation of service notifications and update of the working procedures & scheduling for execution)		Complete

Table A.1-1: *Ageing Management LTO Preparation Activities*

<table>
<tr><th colspan="2">Activity Title</th><th>Activity Description</th><th>Comment</th><th>Completion Date</th></tr>
<tr><td colspan="5">Electrical - Existing Ageing Management Programme Updates</td></tr>
<tr><td colspan="2">Cable ageing management programme (CAMP)</td><td>• Implementation of updated AMP requirements on SAP (creation of service notifications and update of the working procedures & scheduling for execution)</td><td></td><td>Complete</td></tr>
<tr><td colspan="2">Equipment qualification programme</td><td>• Implementation of updated AMP requirements on SAP (creation of service notifications and update of the working procedures & scheduling for execution)</td><td></td><td>Complete</td></tr>
<tr><td colspan="2">Technological Obsolescence Programme</td><td>• Implementation of the proactive Technological Obsolescence Programme.</td><td></td><td>Complete</td></tr>
<tr><td colspan="5">Electrical - One-Time Inspections</td></tr>
<tr><td rowspan="2">One-time inspection (internal and external visual inspection) of electrical cabinets and control boxes on a sample basis</td><td rowspan="2">Sampled scope including electrical cabinets (AR) and control boxes (CR)</td><td rowspan="2">• Perform inspection, evaluate the results, and raise appropriate actions, if required, to address any findings.</td><td>Unit 1 inspections and evaluations performed in Outage 126.</td><td>Complete</td></tr>
<tr><td>Outstanding inspections and evaluations for unit 2.</td><td>Outage 226</td></tr>
<tr><td>One-time inspection (internal visual inspection) to identify the presence of whiskers on the electronic</td><td>Sampled scope including Electronic Cards (ZO, CA, XU)</td><td>• Perform inspection, evaluate the results, and raise</td><td>Unit 1 inspections and evaluations performed in Outage 126.</td><td>Complete</td></tr>
</table>

Table A.1-1: *Ageing Management LTO Preparation Activities*

Activity Title		Activity Description	Comment	Completion Date
components, printed circuit boards and transmitters	and Transmitters (MN, MD, MP)	appropriate actions, if required, to address any findings.	Outstanding inspections and evaluations for unit 2.	Outage 226
Civil Ageing Management Issues – Ageing of Aseismic Bearings				
Ageing and Testing of Aseismic Bearings	Aseismic Bearings	• Perform characterisation of the aseismic bearings to confirm behaviour.		Complete
Civil – New Ageing Management Programmes				
Civil Ageing Management Programme Requirements Manual (CAMPRM)		• Develop and implement an AMP		Complete
Non-metallic liners		• Develop and implement an AMP		Complete
Ageing management programme for aseismic bearings		• Develop and implement an AMP.		Complete
Civil – Existing Ageing Management Programme Updates				
Civil monitoring programme		• Update and implement AMP		Complete

Table A.1-1: *Ageing Management LTO Preparation Activities*

Activity Title	Activity Description	Comment	Completion Date
Civil – One-Time Inspection			
Perform a one-time inspection of the fibre-reinforced polymer vent stack to ascertain the condition w.r.t ageing effects, change in material properties, cracking due to UV radiation, loss of mechanical properties due to elevated temperature, oxygen, and alternating loads or ozone. The results of the one-time inspection will inform further action 1) Auxiliary buildings chimney fibre-reinforced polymer room outdoor	• Perform inspection of 9 HNA 000 BG, evaluate the results, and raise appropriate actions, if required, to address any findings		Complete
To check for soundness, perform a one-time inspection of the post-installed threaded fasteners, bolts, studs, nuts, washers, and screws (without ECS code). Refer to the following commodity spreadsheets attached for locations. 1) Auxiliary buildings, any civil room controlled 2) Auxiliary buildings, any civil room indoor temp hot 35°C	Perform inspections of the following, evaluate the results, and raise appropriate actions, if required, to address any findings • 38-1HKA000BG • 38-2HKA000BG • 38-9HLX000BG • 38-9HNA000BG • 38-2HLX000BG • 38-1HLX000BG • 38-1HWX000BG • 38-2HWX000BG		Complete

Table A.1-1: *Ageing Management LTO Preparation Activities*

Activity Title	Activity Description	Comment	Completion Date
Perform a one-time inspection of 5% of embedded frames linked to ageing commodities (33, 33a, 34, 34a, 35 and 36) for general corrosion, chloride-induced corrosion, and the effect of "loss of material" which will inform any additional action to be taken. Refer to the following commodity spreadsheets attached for locations. 1) Auxiliary buildings penetrations room controlled 2) Auxiliary building penetrations room indoor temp hot 35°C 3) Auxiliary buildings penetrations room indoor uncontrolled 4) Auxiliary buildings penetrations room outdoor 5) Any civil penetrations room controlled 6) Any civil penetrations room indoor temp hot 35°C 7) Anti-seismic area penetrations room outdoor	Perform inspections of the following, evaluate the results, and raise appropriate actions, if required, to address any findings. • 38-1HLX000BG • 38-2HLX000BG • 38-9HLX000BG • 38-1HKA000BG • 38-2HKA000BG • 38-9HNA000BG • 38-1HRX000BG • 38-2HRX000BG • 38-6HQB000BG • 38-1HDA000BG • 38-2HDA000BG • 38-1HDB000BG • 38-2HDB000BG		Complete
Perform one-time inspections of the post-installed threaded fasteners like bolts, studs, nuts, washers, and screws (without ECS codes) to check for soundness/integrity	Perform inspections of the following, evaluate the results, and raise appropriate actions, if required, to address any findings • 38-1HRX000BG • 38-2HRX000BG		Complete

Table A.1-1: *Ageing Management LTO Preparation Activities*

Activity Title	Activity Description	Comment	Completion Date
Perform a one-time inspection of bolting and supports of the pre-cast concrete elements to check for soundness/integrity. Refer to the following commodity spreadsheets attached for locations. 1) Containment building structures concrete room controlled 2) Containment building structures concrete room indoor temp hot 35°C	Perform inspections, evaluate the results, and raise appropriate actions, if required, to address any findings • 38-1HRX000BG • 38-2HRX000BG		Complete
Perform a one-time inspection of bolting and supports of the pre-cast concrete elements to check for soundness/integrity,	• 38-1HRX000BG • 38-2HRX000BG		Complete
Civil – Safety Assessment			
Safety Assessment for the LLWB			Complete

[2]

[2] Valcor solenoid valves and RRA 005 and 007 MTs were initially included in the LTO IPP. These meet the requirements prior to entry into LTO and were therefore removed from Table A.1-1.

Table A.1-2: *LTO Preparation Activities – DSSR and Interim Seismic Hazard Analysis Strategy*

Activity Title	Comment	Completion Date
Submit updated DSSR	DSSR in Progress.	Mar 24
Perform the masonry wall hardening modification identified in the interim SH	Design and installation safety case is complete.	Jul 24
Tsunami probabilistic analysis (return frequency study)	A contract is in place and Tsunami studies are currently in progress.	Mar 24

Table A.1-3: *TLAAs Reanalysed and Processed for NNR Submission*

Item	IGALL Reference	Description of TLAA	Components	Completion Date
1	106	Environmentally Assisted Fatigue	Pressuriser	Complete
2	Additional TLAA	Reactor Pressure Vessel PWSCC	Reactor pressure vessel	
3	Additional TLAA	Reactor Pressure Vessel	Reactor pressure vessel	
4	112	Reactor Coolant Pump Flywheel	Reactor coolant pump flywheel	
5	201	Equipment Qualification	IC in-core thermocouples Cables	
			Rotork Valve Actuators	
			Valcor Solenoid Valves	
			EBA AMRI Actuators - Type C AMRI	
			Jeumont-Schneider	
6	108	Polar Crane	Polar Crane	
7	301	Containment	Strain Gauges	
			Dynamometers	

Item	IGALL Reference	Description of TLAA	Components	Completion Date
			Pendulums	
			Temperature Gauges	
			Concrete Inspections and Repairs	

Table A.1-4: *Knowledge Management*

Activity Title	Activity Description	Completion Date
Organisation and Management		
KM Implementation at Nuclear Engineering		Jul 24
KM Implementation at Operating Department		Jul 24
KM Implementation at Chemistry		Jul 24
KM Implementation at Maintenance		Jul 24
KM Implementation at Radiation Protection		Jul 24
KM Implementation at NPM		Jul 24

A.2 Activities After LTO Implementation

Table A.2-1: *Activities After LTO Implementation – Ageing Management*

Activity Title		Activity Description	Comment	Completion Date
Civil – Containment Structure				
Containment monitoring instrumentation (Linked to IAEA mission finding – Issue area E2)			Modification to be presented at MRC.	Outages 129 and 229
Containment Integrated Leak Rate Testing		Perform containment ILRT test to confirm containment structural integrity.	On track for the scheduled outages. Contract is in place.	Outages 127 and 227
Impressed Current Cathodic Protection (ICCP)			The mock-up for the modification is complete. Modification is on Track.	2025
Electrical				
Equipment qualification – RIC Thermocouples (TC) and TC connectors and cables on Unit 1	1 RIC 001 MT – 051 MT	Replace the RIC cables and associated connector on Unit 1 with qualified cables.	SAP notification is raised for outage 127. Detailed design is approved by the NNR.	Outage 127
PZR heater replacement – Unit 1	1 RCP 005 and 006 RS	Implement modification 13028 (EZP) – replacement of pressuriser heaters 1 RCP 005 and 1 RCP 006 RS	Project is on track in accordance with the schedule.	Outage 127
PZR heater replacement – Unit 2	2 RCP 005 and 006 RS	Implement modification 13028 (EZP) – replacement of pressuriser heaters	Project is on track in accordance with the schedule.	Outage 227

Table A.2-1: *Activities After LTO Implementation – Ageing Management*

Activity Title		Activity Description	Comment	Completion Date
		2 RCP 005 and 2 RCP 006 RS		
Containment C&I penetrations – Unit 1	1 EPP 513, 515, 520, 521, 529 and 530 TW	Implement modification 16001 (RCP) – replacement of the electrical penetrations on unit 1	Project is on track for implementation in outage 127.	Outage 127
Containment C&I penetrations – Unit 2	2 EPP 513, 515, 520, 521, 529 and 530 TW	Implement modification 16001 (RCP) – replacement of the electrical penetrations on unit 2 (2 EPP 513, 515, 520, 521, 529 and 530 TW)	Project is on track for implementation in outage 227.	Outage 227
Refurbish SG snubbers on Unit 2	Snubbers for: 2 RCP 001 GV 2 RCP 002 GV 2 RCP 003 GV	SAP notification raised for refurbishment for the current snubbers	Project is on track for implementation in outage 227.	Outage 227

Table A.2-2: *Activities After LTO Implementation – PSR IIP Safety Improvements*

Activity Title	Completion Date
All PSR IIP Safety Improvements, excluding H1 discussed in Appendix A.1.	Refer to PSR IIP for planned completion dates
Improve the control room envelope (CRE).	Refer to PSR IIP for planned completion dates

Table A.2-3*:* *Activities After LTO Implementation – Interim Seismic Hazard Analysis Strategy*

Activity Title	Comment	Completion Date
Hardened Water External Connection Points (Modification 12004)	Compilation of concept design is in progress. Procurement of long lead items in progress.	Online Oct 25 Outages 127 and 227
Hardened Water Supply (Modification 12008)	Design and procurement in progress.	Jan 25
Install Primary Pump Shutdown Seals (Modification 12023)	Spares for both units are on site. Contract in place for installation during outage 226.	Outages 127 and 227
Plant assessments resulting from the outcomes of the SHA studies.	Studies in progress.	Date to be negotiated with the Regulator

Appendix B
SAR Updates for LTO

This table extracts and consolidates the SAR-specific updates in Appendix A.1, (*LTO Integrated Preparation Plan*).

Table B-1: *LTO Preparation Activities – SAR Updates*

Activity Title	SAR Chapter	Date
Decommissioning description	I-1.2 I-4.0 I-7.0	Complete
Update of activity migration model as well as AADQs	II-5.1 II-5.2 II-5.3 III-4.1 III-5.1 III-5.3	Dec 23
Low-level waste building inclusion.	I-3.0	Jun 23
Update the SAR with the new site characteristics description.	I-2.0	Mar 24
Update the SAR with the description of ageing management	I-4.0 I-4.4.1 I-4.4.2 I-4.4.2.1 I-4.4.2.2 I-4.4.2.2.1 I-4.4.2.2.2 I-4.4.2.2.3 I-4.4.2.2.4 I-4.4.2.3 I-4.4.3	Complete

Table B-1: *LTO Preparation Activities – SAR Updates*

Activity Title	SAR Chapter	Date
Update SAR with a list of updated TLAAs	I-4.0	Complete
Mechanical TLAA Updates		
EAF – Reactor coolant pumps	II-3.1.2.3 II-3.3.2 II-3.3.2.1 II-3.3.2.2 II-3.3.2.4 II-3.3.2.4.2 II-3.3.2.6	Jan 24
EAF – Reactor pressure vessel internals	II-3.1.2.1 II-3.3.1 II-3.3.1.5 II-3.3.1.11	Jan 24
EAF – Pressuriser	II-3.3.5 II-3.3.5.1.1 II-3.3.5.1.2 II-3.3.5.1.4 II-3.3.5.3 II-3.3.5.4 II-3.3.5.10 II-3.3.5.10.1 II-3.3.5.10.2	Jan 24

Table B-1: *LTO Preparation Activities – SAR Updates*

Activity Title	SAR Chapter	Date
EAF – Main coolant lines	II-3.3.4 II-3.3.4.1 II-3.3.4.3 II-3.3.4.5 II-3.3.4.11 II-3.3.4.11.1	Jan 24
EAF – Auxiliary lines	II-3.I II-3.1.2.4 II-3.3.5 II-3.4 II-4.3 II-5.1 II-5.1.2.1 II-7.1	Jan 24
EAF – Control rod drive mechanism	II-3.3.7 II-3.3.7.1 II-3.3.7.3 II-3.3.7.5	Jan 24
EAF – Pressuriser heater sleeves (extra)	II-3.3.5 II-3.3.5.1.1 II-3.3.5.1.2 II-3.3.5.1.4 II-3.3.5.3 II-3.3.5.4 II-3.3.5.10 II-3.3.5.10.1 II-3.3.5.10.2	Jan 24

Table B-1: *LTO Preparation Activities – SAR Updates*

Activity Title	SAR Chapter	Date
Pressuriser spray nozzles – Crack growth analysis of flaws detected in service –AND–Fatigue and thermal ageing analysis of manufacturing flaws and flow tolerance	II-3.3.5 II-3.3.5.1.1 II-3.3.5.1.2 II-3.3.5.1.4 II-3.3.5.3 II-3.3.5.4 II-3.3.5.10 II-3.3.5.10.1 II-3.3.5.10.2	Jan 24
Reactor pressure vessel PWSCC	II-3.1.2.1 II-3.3.1 II-3.3.1.5 II-3.3.1.11	Complete
Reactor pressure vessel wear of adaptor flanges – CRDM pressure housing assembly stress report	II-3.1.2.1 II-3.3.1 II-3.3.1.5 II-3.3.1.11	Complete
Reactor pressure vessel internals – thermal ageing and neutron embrittlement	II-3.1.2.1 II-3.3.1 II-3.3.1.5 II-3.3.1.11	Complete
Reactor pressure vessel internals – vibration	II-2.5.2.4	Complete

Table B-1: *LTO Preparation Activities – SAR Updates*

Activity Title	SAR Chapter	Date
Reactor coolant pump flywheel	II-3.1.2.3 II-3.3.2 II-3.3.2.1 II-3.3.2.2 II-3.3.2.4 II-3.3.2.4.2 II-3.3.2.6 II-3.3.2.6.6	Complete
Electrical TLAA Updates		
Equipment Qualification – Qualified life of EQ components and preservation of qualification	II-1.11	Complete
Civil TLAA Updates		
Polar crane – fatigue of cranes	II-8.5 II-8.5.1 II-8.5.2 II-8.5.3 II-8.5.5	Complete
Reanalysis of the Koeberg containment	II-1.9.2.5.2.1 II-1.9.2.8 II-4.2.2	Complete

Appendix C
Safety-Related Ageing Programmes

Table C-1: *Demonstration for Meeting RG-0027 Requirements for Safety-related AMPs*

RG-0027 Requirement	How it is met	Comments/ Reference
In-Service Inspection Programmes		
i) In-service inspection programmes should be in place and properly implemented for ageing management and evaluations for LTO of applicable in-scope SSCs, including consideration of baseline data.	The In-Service Inspection Program (ISIP) constitutes ISI and IST In service inspection programmes which are both implemented at KOU. In short ISIP= ISI+IST hereunder will refer to ISIP for both. ISI and IST programmes cater for ageing effects in accordance with the 9 IGALL attributes, this requirement is stipulated and called for in the programme engineer's guide which states" *Understanding, preventing, detecting, monitoring, and mitigating a specific ageing effect or degradation mechanism identified on Systems, Structures and Components (SSCs*). The ISI provide in service inspection requirements for safety related static components such as RPV, SG, PZR, Welds, supports and Attachments. The IST provides testing requirements for dynamic components such as valves, pumps, snubbers and the EDG.	ISIPRM: In service Inspection Program Requirements Manual: 240-119362012 ISTPRM: Inservice Testing Requirements Program Manual: 240-97087308 Programme Engineer's Guide 331-148
ii) In-service inspection procedures should be effective in detecting degradation and it should be demonstrated that ageing effects will be adequately detected with the proposed inspection or monitoring technique.	For every inspection/surveillance the ISIP implementers develop working procedures to allow for the implementation of the different ISIP requirements. These procedures are reviewed for adequacy to meet the stipulated requirements and should be accepted / endorsed by the relevant program owners before implementation. The main intent of the ISIP surveillances is to confirm operability readiness and the absence of degradation that could challenge operability of SSCs with safety functions. The monitoring of degradation encompasses all types; namely in service induced, design deficiency and all ageing effects.	KWR-IST-001 (all IST working procedures referenced) KAR-240: The Qualification and Certification of Inspection and Test NDT Personnel and its Sub-Contractors KAR-020: Authorisation of Inspection & Test NDT Personnel and its Subcontractors

Table C-1: *Demonstration for Meeting RG-0027 Requirements for Safety-related AMPs*

RG-0027 Requirement	How it is met	Comments/ Reference
	For the ISI, the NDT working procedures are also reviewed for inspection qualification adequacy where the technique, equipment and the personnel are assessed to confirm capability to detect relevant indications (defects).	240-123597661 (ex-KSA-037): Non-Destructive Testing – Personnel Certification Requirements 240-123588530 (ex-KSA-118): Non-Destructive Testing – Qualification of NDT Systems KGT-047: I&T Training Guide
iii) The results of in-service inspection should be documented such that a trending analysis can be carried out using the results obtained from sequential inspections at the same location and should inform revision of inspection frequencies.	All ISIP inspection and testing results are documented on hard copy data sheets and electronically in a dedicated data base. This record keeping is a statutory requirement driven by the base ASME codes used to derive the ISIP requirements as well as dictated by regulatory commitment. This regulatory requirement is stipulated in the ISIP development standard (Ex KSA-021). Furthermore, this is manged a KOU by the record management system (TD&RM)	IST DB ISI DB (Ideal US based DB) ISIP Vault (hard copies storage facility) 240-110745414 (Ex KSA-021) 331-3
iv) In-service inspection results that indicate notable degradation should be evaluated to ensure that the extent of degradation at similar locations is appropriately determined. SSCs in redundant subsystems should be inspected independently to detect possible differences in their ageing behaviour.	The evaluation process requirements are clearly defined in the ISIP manuals. As an example, the scope expansion/extension is called for by the different ISIP requirements to determine if the found degradation / indication is of a generic nature. The scope selection for ISIP is across the different redundant trains therefore any ageing effect will be detected timeously. In general, the results of ISIP examinations and tests shall be compared to the acceptance standards given in the ISIPRM and ISTPRM. Components where the acceptance standards are not met or where conditions are found which compromise structural integrity or safety function, shall be the subject of an Engineering Work Request (EWR). The evaluation of the reported condition	KAA-688 ISIPRM: In service Inspection Program Requirements Manual: 240-119362012 ISTPRM: Inservice Testing Requirements Program Manual: 240-97087308 Programme Engineer's Guide 331-148

Table C-1: *Demonstration for Meeting RG-0027 Requirements for Safety-related AMPs*

RG-0027 Requirement	How it is met	Comments/ Reference
	shall establish whether corrective action is required or shall justify continued operation without repair / replacement. In accordance with KAA-688 requirements for an evaluation, an extent of condition is always required as part of the investigation.	
v) A list or database should be developed and maintained to document the adequacy of non-destructive examination in detecting, characterising, and trending the degradation of structures or components. The database should provide the technical bases to support the findings and the conclusions necessary to support ageing management decisions.	The ISIP requirement calls for a Database containing the full scope of examinations and tests to be conducted during the Interval and detail inspection plans and schedules in accordance with the requirements of the ISIPRM and ISTPRM. Database controls shall be established to ensure the accuracy, timeliness, and currency of information. The test results are communicated with the relevant department to evaluate the association of these results with possible ageing effects and subsequently decide on the effective measures to be deployed to remedy their negative impact. Practically this could mean a change in the task frequency and requirement under a PM strategy or in some cases a call for a replacement and/or equivalency study.	IST DB ISI DB (Ideal US based DB)
Surveillance Programmes		
i) Surveillance programmes, including functional tests, should be in place and properly implemented for ageing management and evaluations for LTO of applicable in-scope SSCs.	Like point i) above, in addition, KOU decided to keep the surveillance program running independently under the OTS requirement. Although there is an overlap of requirements between the ISTPRM and SRSM however there are benefits in both hence the implementation of both programs concurrently.	ISIPRM: In service Inspection Program Requirements Manual: 240-119362012 ISTPRM: Inservice Testing Requirements Program Manual: 240-97087308 SRSM: KBA-0022-SRSM-000
ii) Particular attention should be paid to the following aspects specifically for nuclear power plants:	All these aspects are specifically addressed by the ISIP manuals and the Containment Leak Rate Testing Program Requirement Manual (CLRTPRM). In brief:	ISTPRM: Inservice Testing Requirements Program Manual: 240-97087308

Table C-1: *Demonstration for Meeting RG-0027 Requirements for Safety-related AMPs*

RG-0027 Requirement	How it is met	Comments/ Reference
(1) The integrity of the barriers between radioactive material and the environment (that is, the primary pressure boundary and the containment); (2) The availability of safety systems such as the reactor protection system, the safety system actuation systems and the safety system support features; (3) The availability of items whose failure could adversely affect nuclear or radiation safety; (4) Functional testing to ensure that the tested SSCs are capable of performing their intended function(s).	• All containment isolation valves (CIVs) are tested every outage under the CLRTPRM. In addition, the Pressure Isolation Valves (PIVs) are tested every outage under the ISTPRM to confirm the pressure boundary integrity of the primary circuit and associated safeguard system • The reactor protection systems are routinely tested under RPR periodic tests. The scope is defined under the ISTPRM, and any ageing effect will be timeously detected. • This is confirmed with the RPR PTs results which are evaluated by engineering. This is performed under the ISTPRM and SRSM • All the above are also driven by the Koeberg Licensing Basis Manual (KLBM) which comprises the complete set of radiation protection and nuclear safety requirements for KNPS, and the principal documentation regarding the processes, programs and practices that demonstrate compliance with these requirements.	SRSM: KBA-0022-SRSM-000
iii) The surveillance programmes should confirm the provisions for safe operation that were considered in the design and assessed in construction and commissioning, and which are verified throughout operation.	This is performed at fixed intervals specified by the ISTPRM and the SRSM. The SRSM criteria are usually derived from the OTS limit to confirm the provisions and assumptions made in the SAR. In general, The ISTPRM and SRSM requirements are conducted to: • Establish a reasonable assurance on operability readiness of components/systems under all design base conditions. Thus, confirming their design base safety function. • Demonstrate operability readiness by confirming the components and/or systems functionality. • Remain within the envelope of hypotheses (assumptions) described in the accident studies of the SAR.	ISTPRM: Inservice Testing Requirements Program Manual: 240-97087308 SRSM: KBA-0022-SRSM-000

Table C-1: *Demonstration for Meeting RG-0027 Requirements for Safety-related AMPs*

RG-0027 Requirement	How it is met	Comments/ Reference
	• Confirm the absence of any unfavourable trends (could challenge operability with time). • Comply with the functional requirements/limits / criteria of the OTS / SRSM.	
iv) The surveillance programmes should continue to supply data from monitoring relevant parameters to be used for assessing the service life of SSCs for the planned period of LTO, for example through existing or additionally installed means for measuring temperature and pressure, or through additional diagnostic systems.	Yes, the surveillance will continue throughout the LTO period. Furthermore, the AMPs identified the need for additional requirements and/or equipment to be used to meet the new requirement when applicable.	ISTPRM: Inservice Testing Requirements Program Manual: 240-97087308 SRSM: KBA-0022-SRSM-000
v) The surveillance programmes should verify that the safety margins for LTO are adequate and provide a high tolerance for anticipated operational occurrences, errors and malfunctions.	This has been confirmed through the SALTO and TLAA studies and analysis. The gaps were identified, and actions were raised where applicable.	
vi) Surveillance programmes using representative material samples (such as material specimens for surveillance of the reactor pressure vessel and cable samples and corrosion coupons) should be reviewed and extended or supplemented for ageing within the period of LTO, if	The reactor pressure vessel material undergoes embrittlement as a result of neutron irradiation over time. A reactor vessel surveillance programme (RVSP) is applied at Koeberg to ensure that the mechanical properties of the RPV beltline materials remain adequate for the LTO of the plant. Verification of RPV beltline material properties is obtained through the surveillance programme by the removal and testing of a series of capsules that contain specimens of the original RPV material. The capsules are removed during	RVSP Manual (under review) See 331-238 (*corrosion*) and 331-127 (*cables*).

Table C-1: *Demonstration for Meeting RG-0027 Requirements for Safety-related AMPs*

RG-0027 Requirement	How it is met	Comments/ Reference
necessary.	outages. Due to fluence lead factors, because the capsules are located closer to the core than the vessel, the embrittlement levels measured on the capsule specimens are representative of the vessel material at times in the future. There are no corrosion coupons currently applied at Koeberg Power Station. Corrosion coupons are generally used to categorise an environment with respect to corrosivity. Currently, Koeberg has implemented a Corrosion Management Programme which is an inspection programme that covers various indoor and outdoor locations throughout the Koeberg site. The programme details are documented in the standard 331-238. Based on walk-downs and visual inspections of the accessible locations of the plant, the inspection results can be used to categorise the corrosivity of the various environments and to take appropriate corrective measures. Koeberg Nuclear Power Station did not install specific cable samples in the plant with the anticipation of performing future ageing studies like other NPP's have done. However, Koeberg Nuclear Power Station will be cutting and removing naturally aged samples from actual functioning cables in the plant to perform ageing assessments during LTO. The cable samples being removed are representative of the cable population being utilised at Koeberg for 1E applications and are removed from the worse anticipated adverse local operating environments. They will then undergo accelerated thermal and radiological ageing after which their nuclear accident resistance and functionality will be checked. This will be a supplement to the further re-analyses that are currently in progress.	

Table C-1: *Demonstration for Meeting RG-0027 Requirements for Safety-related AMPs*

RG-0027 Requirement	How it is met	Comments/ Reference
vii) The documentation on the relevant initial conditions of the material samples used for surveillance should be identified, the adequacy of the information should be assessed, and the documentation should be supplemented as necessary.	Refer to the RVSP manual. See (vi) above related to corrosion and electrical cables.	
viii) Appropriate testing procedures and evaluation methods should be considered for defining the set of specimens to be included in the supplementary material surveillance programme for the reactor pressure vessel, if necessary, at least for alternative assessments such as the master curve approach for assessing fracture toughness.	Refer to the RVSP manual.	

Appendix D
Defence-in-Depth

D.1 Implementation of Defence-in-Depth (DiD)

Defence-in-depth (DiD) is an approach to designing and operating nuclear facilities that prevents and mitigates nuclear accidents. DiD provides multiple, independent and redundant layers of defence to compensate for potential human, organisational and technical failures so that no single layer, no matter how robust, is exclusively relied upon [248].

During the PSR, DiD aspects were assessed in various safety factors, including plant design [63], deterministic safety analysis [64], probabilistic safety assessment [65], hazard analysis [66], and emergency plan [72] reviews. The impact of deviations on the levels of DiD and the fundamental safety functions (FSFs), identified during the PSR, were assessed in the PSR global assessment (GA) [115].

This section utilises the outcomes of the PSR to demonstrate the adequacy of the levels of DiD at Koeberg to support LTO.

D.1.1 Plant Design Provisions for Defence-in-depth

The PSR review has adequately assessed the provisions available in the plant design at each level of DiD to confirm that the objectives of the levels of defence-in-depth are met. The plant design safety factor demonstrated that the DiD concept is embedded at Koeberg [63]. The SAR provides details on the approach to DiD at Koeberg and identifies three levels of DiD as follows:

- **Level 1** – Prevention of abnormal operation through conservative design and quality of fabrication.
- **Level 2** – Control of abnormal operations through the provision of protection systems to prevent accidents and allow safe shutdown.
- **Level 3** – Control of accidents which may affect the integrity of the fission product barriers through the provision of design features to protect the public.

The SAR further notes that a “Level 4” defence-in-depth also exists at Koeberg, which provides for the mitigation of radiological consequences in the event of radioactive releases through the implementation of an emergency plan and emergency operating procedures.

The concept of DiD has evolved given the benefit of operational experience. IAEA INSAG 10 (*Defence-in-Depth in Nuclear Safety*) [248] is based on five Levels of DiD as follows:

- Level 1: Prevention of abnormal operation and failures by design.
- Level 2: Prevention and control of abnormal operation and detection of failures.
- Level 3: Control of faults within the design basis to protect against escalation to an accident.

- Level 4: Control of severe plant conditions in which the design basis may be exceeded, including protecting against further fault escalation and mitigation of the consequences of severe accidents.
- Level 5: Mitigation of radiological consequences of significant releases of radioactive material.

The application of DiD at Koeberg does not fully align with IAEA INSAG 10 (*Defence-in-Depth in Nuclear Safety*) [248], however, Koeberg has available provisions for complementary accidents and a full set of severe accident management guidelines which aims to satisfy the requirements of Level 4 DiD. The 'Integrated Koeberg Nuclear Emergency Plan (IKNEP)' (KAA-811) [148] is authorised and exercised regularly and serves as Level 5 of DiD. The emergency plan is discussed in § 9.7.2. Safety improvements have been included in the PSR IIP to fully align Koeberg with modern practice in terms of the DiD concept.

The plant design PSR confirmed that operating practices at Koeberg ensure that existing provisions at each level of DiD are kept available (that is, a level of defence is not optional but a necessary safety feature) and where this is not the case, suitable justification is provided. Koeberg can address deviations from normal operations and control the plant within design limits. This operating philosophy is embedded in the operating technical specifications which ensures that the plant is operated within design limits.

The PSR PSA review [65] confirmed that the plant's design is balanced. No particular feature or postulated initiating event makes a disproportionately large or significantly uncertain contribution to the core damage and large early-release frequencies. During this review, it was also confirmed that the levels of DiD are independent. The PSA review concluded that Koeberg has sufficient plant procedures and equipment to prevent design-based accidents and prevent or mitigate severe accidents.

The FSFs are ensured through levels of DiD which serves to protect the physical barriers, namely the fuel cladding, reactor coolant system boundary and the containment. The Koeberg design makes provision for the protection of these barriers which are discussed further in § 9.4.2. The FSFs are discussed in § 9.4.5.

D.1.2 Independence of the Levels of Defence-in-Depth

The deterministic safety analysis [64] and plant design [63] safety factor reviews assessed whether the SSCs required for implementing safety functions at any one level of DiD are sufficiently independent of those at other levels of DiD, considering the threats that can affect them.

The PSR plant design review found that the redundant parts of a system performing safety functions were physically separated from each other. It was determined that interference between safety systems or between redundant or diverse elements of a system is prevented by various means, including electrical isolation and physical barriers.

Furthermore, it was found that generally, SSCs assigned to different levels of defence in depth are functionally isolated from one another. This ensures that the mode of operation or the failure of a system or component of a lower level does not result in the malfunction or loss of function of a system of a higher level. Similarly, the failure of higher-level SSCs does not impair the function of lower-level systems. For example, the auxiliary feedwater system is functionally isolated (physically separate, including separate power supplies, water sources and flow paths) from the main feedwater system and failure of the main feedwater system will not result in failure of the auxiliary feedwater system.

There are however examples where independence between safety functions is not achieved. For example, the containment spray system (EAS) and safety injection system (RIS) pumps are subject to common-cause failures, as they are both cooled by the component cooling system (RRI) and essential service water systems (SEC), which also cools the EAS heat exchangers and therefore cannot be considered independent. However, in accordance with the general DiD expectations, independence between physical barriers is necessary "as far as reasonably practical" [296]. There are specific cases when the independence of barriers is not possible to maintain. For Koeberg, this is such a case. To reduce the risk, physical separation of redundant trains of RRI/SEC, JPC backup to EAS, limiting conditions of operation and incident procedures for the total loss of RRI/SEC provide some DiD, together with implementation of human performance tools.

The PSR plant design review, supported by the PSA and DSA reviews has adequately demonstrated that there is sufficient independence between the levels of DiD such that failure at one level or barrier of defence does not cause the failure of others, to the extent practical. A strength related to the design of equipment takes into consideration diversity, redundancy, physical separation, and functional independence was noted during the PSR. Another strength is related to the single-failure criterion which is accounted for within the inherent design of the SSCs at Koeberg. These strengths demonstrate that the DiD concept is well embedded at Koeberg and no single human, organisational or technical failure could lead to harmful radiological effects.

D.1.3 Equipment Qualification in Defence-in-Depth

Equipment qualification (EQ) is an important design attribute used to minimise common cause failures of items important to safety due to issues related to functional, seismic, and environmental capability, potential electromagnetic interference, and harsh environmental conditions over the full range from normal operation, anticipated operational occurrences, and accident conditions. According to the outcome of the PSR EQ review, the plant has implemented a formal EQ programme to ensure that qualified equipment important for nuclear safety, located in harsh plant environments, is qualified to perform its safety function throughout specified plant states. The EQ programme takes into consideration the effects of in-service ageing. The EQ programme processes and procedures were found to be aligned with international standards.

However, environmental parameters and service conditions applicable to DECs have not been fully derived and incorporated into the EQ programme. Safety improvements to address this gap are included in the PSR IIP. Qualified equipment is maintained and preserved through good maintenance, inspection and testing regimes. Implementation of the safety improvements will ensure that equipment important to safety is properly qualified to perform their intended safety functions in all anticipated operational occurrences and accident conditions.

The EQ programme is continually reviewed to ensure that equipment important to-safety is qualified for the duration of LTO. The EQ programme is discussed further in § 9.5.1.5.1.

D.1.4 Practical Elimination of Significant Radioactive Releases through Defence-in-Depth

The concept and safety demonstration of 'practical elimination' of early and large releases are interpreted in the IAEA SSR 2/1 (*Safety of Nuclear Power Plants: Design*) [266] and IAEA-TECDOC-1791 (*Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants*) [272] as:

- Physical Impossibility for the accident sequence to occur, or
- If the accident sequence can be considered with a high degree of confidence to be extremely unlikely to arise.

This deterministic concept of 'practical elimination' is a fairly new requirement aimed mainly at new plant design. The NNR specify principal safety criteria in RD-0024 (*Requirements on Risk Assessment and Compliance with Principal Safety Criteria for Nuclear Installations*) [289] that include probabilistic public risk limits and a bias towards accidents with early and large releases. Consequently, the PSA (both Level 1 and 2) largely considers the scenarios described in IAEA-TECDOC-1791. Therefore, provisions have been implemented and the PSA demonstrates that these scenarios are unlikely and the principal safety criteria are met.

IAEA-TECDOC-1791 [272] notes that for new designs which adopt the latest technological solutions for a strong implementation of defence in depth, it is expected that a large or early release frequency below 1E-6 per reactor year could be achieved for internal events. More demanding requirements are imposed for protection against external events.

The following severe accident conditions are described in TECDOC-1791 (*Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants*) [272] and should be considered for practical elimination:

D.1.4.1 Events that could lead to prompt reactor core damage and consequently early containment failure

1. Failure of a large component in the reactor coolant system (RCS)
   - ∗ Koeberg provisions to prevent or mitigate this event include:
     - The RCS system at Koeberg was built in accordance with ASME and US NRC 10 CFR 50 (General Design Criteria) codes and standards.

- Quality control during the manufacturing and construction phase and pre-service inspections were conducted in accordance with well-established codes and standards.
- Koeberg has test samples in the RPV to assess neutron embrittlement of the vessel.
- ORT (operation at reduced temperature) was implemented at Koeberg which reduced the primary side temperature. This minimized the effects of SCC (stress corrosion cracking). Subsequent to the implementation of the new SGs, ORT will be stopped.
- Major RCS components are physically separated and appropriately protected from missiles from inside containment and outside containment.
- Pipe breaks in the RCS system are analyzed in the SAR with design and operational provisions to monitor RCS leakage and limit the consequence of loss of coolant.
- The RPV head replacement addresses the effects of ageing of the RPV head.
- Implementation of comprehensive in-service inspection and maintenance programmes in accordance with international codes and standards, including the integration of OE.

∗ Planned Plant Improvements:

- The steam generators (SGs) are due for replacement prior to LTO.

2. Uncontrolled reactivity accidents

∗ Koeberg provisions to prevent or mitigate this event include:

- The core has negative reactivity coefficients that remain negative with all combinations of reactor power, pressure and temperature. (SAR II-2.4.5.1 (*Core Reactivity Control*) and III-4.3.2.4.3 (*Accident Analysis*) [178])
- An anti-dilution protection modification, based on an EDF design, has been installed as part of the CP1 modifications.
- On loss of forced cooling, when RRA is not connected to the RCP, RCV charging suction is swapped to the borated PTR tank.
- The SBO seal injection modification takes suction from PTR tanks.
- Dilution related equipment failures and valve positions actuate control room alarms (REA 531, 533, 535 AA).
- Shutdown control rod banks are withdrawn and maintained in the withdrawn position during shutdown operations.
- Reactor flux set-point during shutdown and for negative reactivity margin during reloading and maintenance shutdown are set to allow the operator sufficient time for action if accidental dilution occurs under these conditions.
- Reactivity insertion accidents are analysed in the SAR as part of the deterministic safety analysis.

D.1.4.2 Severe Accident Phenomena which could lead to Early Containment Failure

The affected safety functions are loss of containment Integrity.

1. Direct containment heating (DCH) by high-pressure core melt

   * Koeberg provisions to prevent or mitigate this event include:

     - SAG-2 instructs the TSC to depressurize the RCS at the onset of core melt using available design provisions for depressurisation, for example, PORVs.
     - The RPV is housed in the reactor pit which provides some protection for the containment wall from ejected corium.
     - The EPRI report "Severe Accident Management Guidance Technical Basis Report" provides a detailed description of the requirement for DCH and concludes that plants with a similar containment layout to Koeberg, DCH was not considered a credible means of containment failure.

2. Large steam explosion

   The affected safety functions are loss of both RCS and containment integrity. The two key concerns are the in-vessel steam explosion and the ex-vessel steam explosion.

   * Koeberg provisions to prevent or mitigate this event include:

     - Koeberg has a dry reactor pit design meaning it is not designed for reactor pit flooding. EAS water may nonetheless enter the reactor pit through the hatches of the RPN detector channels.
     - The EPRI report, 'Severe Accident Management Guidance Technical Basis Report' considers multiple events such as the catastrophic collapse of the core debris into the water remaining in the lower plenum and ejection of the RPV head as a missile with sufficient velocity to fail the containment wall upon impact. It concluded that in-vessel steam explosion has a low probability of occurrence.
     - From the analyses used as input into the Level 2 PSA, it can be concluded that the probability of an ex-vessel steam explosion failing containment integrity is generally considered negligible.
     - In the transients where the containment spray system is not functioning (at least until after vessel failure), there is insufficient water in the reactor pit at the time of vessel failure for there to be any significant chance of a steam explosion failing containment.
     - For scenarios where there is water accumulated in the reactor cavity there is a remote chance of an ex-vessel steam explosion, despite one occurring, it is unlikely to fail containment integrity.
     - In the shutdown severe accident transient analyses, the eight neutron detector covers will be in place and subsequently the reactor pit is expected to be dry.

3. Hydrogen detonation

   The affected safety functions are loss of both RCS and containment integrity. The two key concerns are in-vessel hydrogen explosion and ex-vessel hydrogen explosion.

   * Koeberg provisions to prevent or mitigate this event include:
     - Koeberg installed passive hydrogen recombiners (PARs), 24 per unit, replicating the EDF CP1 plant design.
     - Koeberg is not able to measure hydrogen concentration in the containment (a computational aid exists for hydrogen flammability in containment).
     - The SAMGs recommend 'steam inerting' as the strategy for managing hydrogen during the first 6 hours after core damage has occurred. This strategy prevents hydrogen from burning and gives the PARs more time to recombine the hydrogen.

D.1.4.3 Severe Accident Phenomena which could lead to Late Containment Failure

1. Molten core concrete interaction (MCCI)

   The primary concern with MCCI is maintaining containment integrity. This is threatened by primarily basement melt-through and the production of additional hydrogen due to the MCCI.

   * Koeberg provisions to prevent or mitigate this event include:
     - Other than the thick basemat, there were no original plant design features installed to deal with MCCI. Thermocouples were installed in the basemat under the reactor (SAR III-3.4.2 (*Surveillance*) [178]), to aid in detecting MCCI.
     - A strategy exists in the SAMGs (SCG-3) to mitigate this concern by flooding containment to ensure that water gets into the reactor pit and cools the corium (taking into consideration potential negative effects of such a strategy).

2. Loss of containment heat removal

   * The affected safety functions are loss of both core cooling and containment integrity.
   * Koeberg provisions to prevent or mitigate this event includes:
     - The EAS containment cooling and spray system provides the primary method for cooling the containment and the core (after a loss of primary inventory).
     - A backup spray system has been installed but does not have any heat/energy removal capability via the firefighting water production system (JPP).
     - The SAMGs provide further strategies for managing containment pressure.
   * Planned plant improvements

- Implementation of the external hardened water connection point modification (EERI 12004) which provides external connection points for mobile pump injection to the EAS spray header injection.
- Implementation of the filtered containment venting modification (EERI 12042).

D.1.4.4 Severe Accident with Containment Bypass

* Koeberg provisions to prevent or mitigate this event includes:
  - Koeberg's design is one of few PWR designs where the residual heat removal system (RRA) is entirely in the containment building.
  - In the SAMG, the SG tube integrity is considered the severe accident guideline with the highest priority.
  - Safety Justification, J2010/0008 confirmed that preventive measures are taken in the Koeberg SG health care programme to minimize SG tubing degradation mechanisms, reducing the probability of an SGTR (steam generator tube rupture) to an acceptably low level.
  - Most containment penetrations pass through the nuclear auxiliary building (NAB), which would provide some scrubbing.
  - Cautions exist in the EOPs and SAMGs for EAS backup to RIS to avoid containment bypass when taking suction from the containment sumps.
  - Breaching of containment integrity during shutdown is controlled through technical specifications and an outage safety plan.
* Planned plant improvements
  - SG replacement modification is scheduled for implementation prior to LTO.
  - Equipment hatch bolting upgrade.
  - The 'Filtered Containment Venting' (EERI 12042) modification. This will prevent containment over pressure by providing containment venting through a filtered system.

D.1.4.5 Significant Fuel Degradation in a Storage Pool

The affected safety functions are spent fuel pool (SFP) cooling and radiological confinement.

- Koeberg provisions to prevent or mitigate this event includes:
  * If cooling is not restored or make-up is not provided, a minimum of 21.85 hours is available before fuel is uncovered (at 12.2 m level with 11.5 MW heat load) [221].

   * Alternative inventory make-up is provided by either the SED, JPP or the mobile fire protection equipment (JPS) systems, each of which can cope with the maximum boil-off rate with sufficient margin and prevent fuel uncover.
   * One PTR pump and one heat exchanger can provide sufficient cooling to prevent boiling even for a maximum heat load of 11.5 MW.
   * Strategies for managing SFP accidents are provided in incident and accident procedures, I-PTR and SAMGs.
   * The SFP has an upgraded fuel loading crane.
   * The unintentional dilution of boron concentration in the SFP is analysed in the SAR (III-4.3.6.5). Criticality in the SFP is prevented by the physical separation of fuel assemblies, by the presence of borated water in the SFP and by the use of neutron absorptions with fixed poisons.
   * Borated water is available from several sources to mitigate a SFP dilution event, including the refuelling water storage tank (PTR 001 BA), reactor boron and water make-up system (REA) and the compartments adjacent to the SFP.
   * Misplacement of fuel in the SFP is analysed in the SAR (III-4.3.6.6). The spent fuel pit bridge has a fuel tracking database system that prevent the physical misplacement of a fuel assembly into a Region II storage rack.
   * Overall fuel damage frequency for misplacement of fuel assemblies, door seal failure, loss of off-site power, loss of SFP inventory due to pipe failure, loss of SFP cooling due to PTR system failure, and boron dilution is estimated as 5,03E-8/unit per year, excluding external hazards [221].
   * A review of the earthquake and tsunami external event EERT-12-024-RPT (*Earthquake and Tsunami with Induced Events*) [130] confirmed that the SFP structural integrity, SFP cooling and SFP emergency makeup (via JPP) remains available following a design basis earthquake and tsunami.
   * However, earthquakes and tsunami more severe than design may result in loss of all cooling and emergency make-up to the SFP. Cooling via bulk boiling can be preserved if sufficient make-up water can be provided as planned by the implementation of modifications 12008 and 12004.

- Planned plant improvements
   * Implementation of EERI modifications for hardened water supply - 12008 and hardened external connection points – 12004 provides adequate provision to get cooling water into spent fuel pool during DEC scenarios and are sufficiently robust to withstand the anticipated DEC.

* Enlarge SFP siphon breaker to cater for guillotine breaks.

D.1.5 Human Factors and Performance Consideration in the Implementation of Defence-in-Depth

Human and organisational aspects are of particular importance to DiD, and these include aspects such as:

- Safety culture;
- Quality assurance and quality processes;
- Maintenance and operations;
- Staffing, competency, and skills; and
- Severe accident management.

Safety culture has been assessed during the management systems safety factor review [69] and is discussed further in § 9.9. It was determined that the nuclear safety culture programme is well established at Koeberg, safety culture is sufficiently monitored, and potential negative trends are detected and corrected. The safety culture at Koeberg is considered acceptable. Koeberg's integrated management system complies with RD-0034 (*Quality and Safety Management Requirements for Nuclear Installations*) [290] and ISO 9001 (*Quality Management Systems Requirements*) [278], so the management system includes audits, independent reviews, quality control inspections and continuous improvement processes to support DiD.

The review determined that maintenance and operations are performed in accordance with documented procedures to ensure consistency and quality outcomes. Critical tasks require peer checking and quality control checks, in addition to self-checking as part of a comprehensive set of human performance tools which provide an essential layer of DiD against human errors.

Staff training was assessed during the human factor's safety review [70]. The review determined that staff are trained and assessed for competency in accordance with internationally benchmarked procedures. This is followed by periodic requalification to reconfirm competency. Human performance tools are integrated into qualification training to ingrain a strong nuclear safety culture.

Severe accident management guidelines are developed and trained on during emergency plan exercises. These guidelines are comprehensive and together with mobile severe accident management equipment, provide good support for level 4 and level 5 DiD.

The human reliability analysis and the extent to which human actions are considered in operating, and accident procedures have been adequately assessed during the probabilistic safety assessment (PSA) safety factor review [65]. The PSA review concluded that human actions to the extent assumed in operating and accident procedures are considered. Plant personnel actions are reflected in the assessment of risk and human reliability analyses are performed taking into consideration the factors which can influence the performance of plant staff.

The DSA safety factor review [64] also assessed the short-term time-critical operator actions during DBA and AOOs. The timing of operator actions (based on simulator and crew training results) was found to be in line with ANSI 58.8 (*Time Response Design Criteria for Safety-Related Operator Design Criteria Actions*) [234].

Based on the above assessments and outcomes, human factors are adequately considered in the implementation of DiD at Koeberg.

D.1.6 Impact of PSR Deviations on Defence-in-Depth – Global Assessment (GA)

The adequacy, acceptability and robustness of the DiD levels at Koeberg were assessed during the PSR GA and documented in appendix C of the GA report [115]. All the deviations identified during the PSR were assessed for their impact on DiD and FSF, which are discussed below.

D.1.6.1 Description of the Defence-in-Depth Assessment Approach

RG-0028 [295] requires that the GA includes a review of the extent to which safety requirements relating to the concept of DiD are fulfilled. INSAG-10 (*Defence-in-Depth in Nuclear Safety*) [248] requires that the safety assessment focuses on possible challenges to levels of defence. An essential element of such an assessment is a judgement of the extent to which the FSFs are ensured through levels of DiD.

The GA DiD analyses were focused on determining the individual and cumulative impact of safety factor deviations on the five levels of DiD and the FSFs to evaluate the adequacy of the existing provisions at Koeberg. The updated IAEA SRS-46 (*Assessment of Defence-in-depth for Nuclear Power Plants*) [260] guide which incorporates the lessons learned from the Fukushima-Daiichi accident, was used to conduct the DiD analyses.

The IAEA SRS-46 (*Assessment of Defence-in-depth for Nuclear Power Plants*) [260] provides a set of objective trees (OT) for each of the INSAG-10 (*Defence-in-Depth in Nuclear Safety*) [248] safety principles belonging to the five levels of DiD. The objective tree is a graphical representation of the objective of the level of DiD, the safety function, the safety function challenges, the mechanisms that constitute the challenge and the provisions to prevent the mechanisms from occurring to protect the safety function (refer to Figure D.1-1).

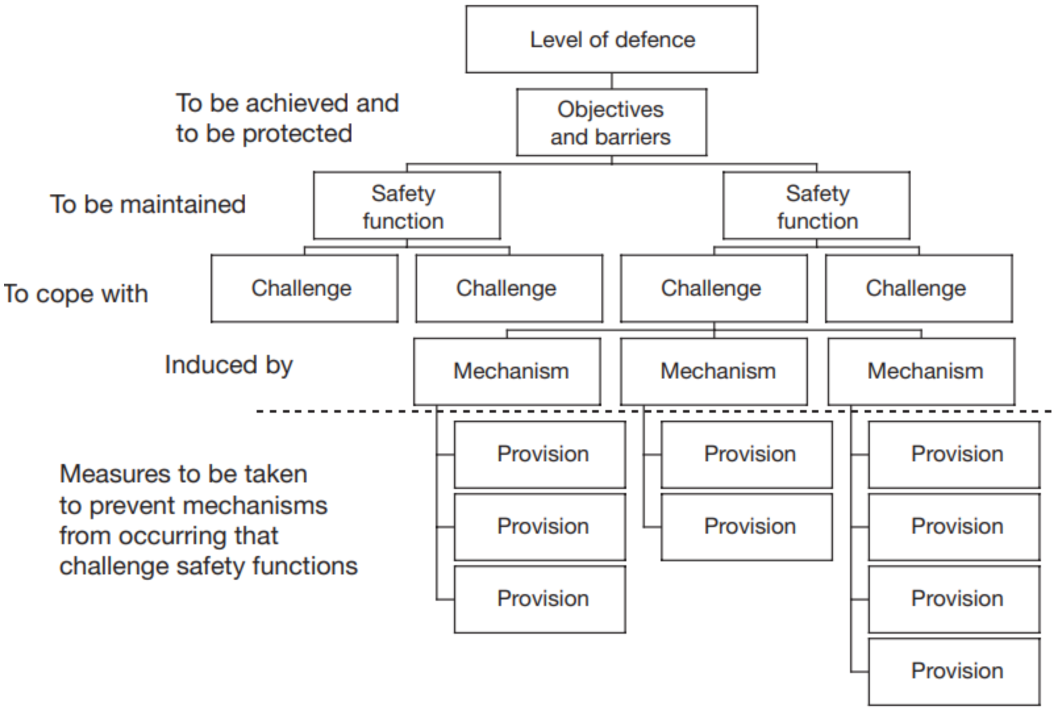

Figure D.1-1: *DiD Objective Tree Structure in IAEA SRS-46 Used in the GA*

The safety significance of deviations is considered in assessing the challenges to safety functions, affected levels of DiD, and available provisions possibly compensating for the deviations. This method indicates the level(s) of DiD affected by each deviation. Five levels of DiD were analysed as defined in INSAG-10 (*Defence-in-Depth in Nuclear Safety*) [248]. Level 4 for design extension conditions (DEC) was divided into two sub-levels in accordance with the IAEA approach, namely Level 4A (DEC-A) and Level 4B (DEC-B). This approach enabled conclusions to be drawn on the extent to which levels of defence are affected by deviations identified during the PSR.

The assessment was carried out by a multidisciplinary team of experts. Below is a summary of the outcomes but with an increased focus on the impact of deviations graded high and medium on each level of DiD.

D.1.6.2 Results of impact Analysis on Each Level of Defence-in-Depth

The PSR deviations identified by each safety factor review and which impacts each of the levels of DiD and FSF are shown in Table D.1-1, noting that a deviation can impact more than one level of DiD and FSF. There is one deviation graded High[3] which impacts levels 2, 3 and 4. The number of deviations graded Medium impacting levels 1, 2, 3, 4 and 5 are one, two, nine, twelve and

[3] A safety justification performed concludes that the control room habitability criterion of 50 mSv is met when taking into account actual measured plant input parameters. Based on the safety justification the initial high risk identified by the PSR has been reduced.

respectively. The deviations graded High and Medium affecting each of the five levels of DiD are discussed in attachment D.2.

Table D.1-1: *Analysis of the Deviations on DiD Levels and FSFs*

Deviations	**Level 1**	**Level 2**	**Level 3**	**Level 4a**[4]	**Level 4b**[5]	**Level 5**	**Reactivity Control**	**Core Cooling**	**Confinement of Radioactive Material**
Total deviations	12	19	40	51	44	30	11	25	21
Deviations graded High	0	1	1	1	1	0	1	1	1
Deviations graded Medium	1	2	9	12	10	3	1	4	1
Deviations graded Low and Drop	11	16	30	38	33	27	9	20	19

Based on the assessment comments provided in attachment D.2, the impact of these deviations on the existing levels of DiD is no more than minimal. The deviations related to the updating of the site safety report and the control room envelope in-leakage will be resolved commensurate with their risk. The implementation of the safety improvements to address these deviations, which are contained in the PSR IIP, will bring Koeberg in line with international standards, and provide significant improvement mainly in the field of DEC. Safe continued operation, including LTO, is therefore justified.

D.1.6.3 Results of Impact Analysis on Defence-in-depth Objective Trees

The potential impact of deviations on 124 different OT branches (mechanisms) was identified which represent more than a third of the total number of OTs and mechanisms as defined in IAEA SRS-46 (*Assessment of Defence-in-depth for Nuclear Power Plants*) [260]. The breakdown is as follows (refer to attachment 1 of Appendix C of the global assessment report) [115]:

- 74 OT branches (mechanisms) were affected by one deviation.
- 42 OT branches were affected by two, three, four or 5 deviations.
- 8 OT branches were affected by more than five deviations and are listed in Table D.1-2.

[4] DEC without Core Melt, EOPs Used

[5] DEC with Core Melt, SAMGs Used

These results agree well with the analyses of the impact on individual levels of DiD and FSFs, as well as the analyses of fundamental safety principles that were analysed separately (by a different group of experts) in Appendix B of the global assessment report [115].

Table D.1-2: *List of OT Branches Affected by More Than Five Deviations*

OT Number (mechanism)	Safety Principle	Challenge	OT branch (mechanism) affected by deviations	No of Deviations	No of high and medium deviations
18_4 (33)	General basis for design	Inadequate design basis for AOOs and accident conditions	Safety analysis of AOO, DBAs and DECs not properly performed	25	1
18_3 (32)	General basis for design	Inadequate design basis for AOOs and accident conditions	Inadequate selection of PIEs and accident scenarios	8	2
18_5 (34)	General basis for design	Inadequate performance of items important to safety	Items important to safety are not properly designed to cope with design bases	8	1
26_2 (64)	Radiation protection in design	Discharges above the prescribed limits	Inadequate control of effluent activity	8	0
23_4 (53)	Dependent failures	Items important to safety fail when performing their functions due to common-cause failure vulnerabilities	CCF due to internal hazards (such as flooding, missiles, pipe whip, jet impact)	7	3
71_5 (265)	Feedback of operating experience	Latent weaknesses in plant safety due to ineffective feedback from operational experience	Unidentified degradation trends in the performance of items important to safety	7	0
24_3 (60)	Equipment qualification	Required reliability not maintained throughout plant lifetime	The ability of SSCs to withstand specified service conditions affected by ageing	6	0
75_3 (285)	Training and procedures for accident management	Inadequate response of AM personnel due to inadequate AM procedures and guidelines	Severe accident management guidelines inadequate	6	1

The most affected challenge (grouping of mechanisms) is the inadequate design basis for AOOs and accident conditions which is linked to the general basis for design safety principle and OT 18_3 and 18_4 each with 8 and 25 deviations respectively. While the number of deviations is high, it was judged that there is no cumulative effect because the deviations collectively do not have an adverse amplification effect on the provisions. The deviations associated with the OT branches in Table D.1-2 and graded High or Medium, namely 1E-19-D1, 5A-01-D1, 5A-04-D1, 5B-08-D1 and 7I-03-D2 are discussed in attachment D.2. Since there is no cumulative effect and the arguments provided in attachment D.2, the impact on the DiD safety function is no more than minimal. However, as noted in § D.1.6.2, the safety improvements included in the IIP to address these deviations will have a significant improvement in Koeberg's capability to respond to DECs as it will bring Koeberg in line with international standards related to DECs.

The impact of all deviations on the affected provisions was assessed as 'Negligible', 'Low', 'Medium', or 'High' (refer to attachment 1 of Appendix C of the global assessment report) [115] (**Note:** The assessment of 'Negligible', 'Low', etc. does not refer to the deviation grading but rather the impact of the deviation on the specific provision). The impact is considered high if the deviation directly contradicts the provision such that the provision is completely not met. Only four deviations were assessed to have a high impact on any OT branch. These were deviations related to the control room in-leakage (1E-30-D1), the design extension condition list not fully derived and justified (1F-07-D1), the outdated site safety report (7A-30-D1) and modifications to address external hazards not completed (7I-03-D1). One deviation (7I-03-D1) had a high-level impact on nine OT branches. The reason was that this deviation covers a suite of outstanding DEC modifications. No OT branches were impacted by more than one deviation with high impact.

No cumulative effects of multiple deviations on OTs were observed, and thus no new global issue was identified. GI-001 (inadequate identification, justification, analysis and documenting of design extension condition events) was identified in the consolidation of PSR findings outcomes that preceded the DiD assessment in the GA.

No new deviations were found based on the OTs DiD impact analyses.

D.1.6.4 Results of Impact Analysis on Fundamental Safety Functions

The fundamental safety functions (FSFs) ensure that the three barriers that prevent the release of radioactivity are protected, so they are essential for the effective performance of DiD. The impact of the PSR deviations on the FSF was assessed during the GA [115]. Attachment D.3 contains a list of deviations graded 'Medium' or 'High' that have a direct impact on the FSFs. As mentioned in § D.1.6.2, the deviations related to the control room envelope and the update of the site safety report will be resolved commensurate with their risk. Based on the arguments provided in Attachment D.3, the PSR deviations do not have a significant impact on the FSFs. The analysis concluded that there is no significant cumulative effect because of the collective impact of the deviations on the FSFs [115].

The FSFs therefore remain capable of protecting the barriers in all applicable plant states, when needed.

D.1.7 Adequacy of Defence-in-Depth at Koeberg

It has been demonstrated that DiD is embedded at Koeberg in its plant design, operations and management systems. It has also been demonstrated that there is sufficient independence between the individual levels of DiD. It has been shown that the deviations identified during the PSR have a limited impact on DiD as adequate compensatory measures exist, there is no significant cumulative effect and several of the deviations will be resolved even prior to LTO. The FSFs can be ensured despite the impact of the deviations as they do not have a cumulative effect on the FSFs and are mainly of a low safety significance. Protection of the physical barriers, namely fuel cladding, reactor coolant system and containment, which is integral to an effective DiD, is extensively considered in the SAR with available design provisions to ensure its protection.

While the above measures provide a good indication of the adequacy of DiD, a risk-based assessment particularly the use of PSA provides a good measure of the adequacy of the DiD [296]. Koeberg complies with the principal safety criteria (risk limits) set by the NNR in terms of core damage frequency, large early release frequency and peak public risk. These are discussed in § 9.2.2.

In conclusion, Koeberg's DiD is deemed adequate for continued safe operation including LTO. Safety improvements, particularly related to DEC (level 4), will further enhance Koeberg's DiD and are included in the LTO improvement plan.

D.2 Impact of Deviations on DiD Levels

Table D.2-1: *Impact of Deviations on DiD Levels*

Deviation Nº	Deviation title	Level 1	Level 2	Level 3	Level 4a[6]	Level 4b[7]	Level 5	Assessment Comments
1B-20-D1	Unidentified Plant Design Shortfalls Leading to Internal Hydrogen Explosion Risk			X	X	X		The Explosion Hazard Report on the risk of an internal hydrogen explosion concluded that "buildings housing safety related equipment required to safely shut down the plant will not experience an overpressure from any source to the point where severe damage is expected." However, some damage may be incurred by the radiators of one of the EDGs. A hydrogen explosion of this type and magnitude is considered to be of a relatively low frequency due to a low record of incidents. As further mitigation, there has been significant progress made in the HAZLOC programme management with the implementation of modifications, processes and procedures aligning with best practice requirements. Safety improvements to address this risk is included in the PSR IIP.

[6] DEC without Core Melt/ EOPs Used

[7] DEC with Core Melt - SAMGs Used

Table D.2-1: *Impact of Deviations on DiD Levels*

Deviation Nº	Deviation title	Level 1	Level 2	Level 3	Level 4a[6]	Level 4b[7]	Level 5	Assessment Comments
1E-18-D1	Accuracy of the flow indicators for balancing of the injection flow on the high-head injection lines to the cold legs is insufficient			X	X	X		The balancing of the injection flow is performed during periodic testing. Three clamp-on ultrasonic flow meters are used to compare the injection flow. For the range of flows, the ultrasonic readings were always within 5% of the indication. The requirement of a balanced flow is therefore assured using an alternative methodology. A safety improvement to address this risk is included in the PSR IIP.
1E-19-D1	Both trains of the component cooling system susceptible to common cause failure during a flooding event			X	X	X		This plant deficiency is already incorporated into the baseline PSA model. Incident procedures for loss of component cooling do exist and some mitigations have been implemented that is, a more visible sump level red alarm. The core damage frequency due to flooding of the pump rooms to be 5.5E-08. A Modification has been raised to deal with this plant deficiency and included in the PSR IIP.
1E-30-D1	Control Room Envelope (CRE) Confinement Questionable Performance		X	X	X	X		Inadequate CRE confinement poses a risk to the control room operators because the dose criterion may be exceeded during certain accident conditions. However, a safety justification performed concludes that the control room habitability criterion of 50 mSv is met when taking into account actual measured plant input parameters. Based on the safety justification the initial high

Table D.2-1: *Impact of Deviations on DiD Levels*

Deviation Nº	Deviation title	Level 1	Level 2	Level 3	Level 4a[6]	Level 4b[7]	Level 5	Assessment Comments
								risk identified by the PSR has been reduced. Nevertheless, several mitigating actions have been successfully implemented, and more are planned to reduce the unfiltered in-leakage into the control room. The improvement actions associated with this deviation will be prioritised commensurate with its risk and are included in the PSR IIP.
5A-01-D1	List of Postulating Initiating Events (PIEs) with respect to international standards is missing, not identified nor justified.			X	X	X		There is a lack of alignment of the KNPS PIEs to international standards. While the list of DBAs postulated initiating events (PIEs) do largely align with international norms, the review did identify some PIEs not considered. When considering DEC, that difference is somewhat larger. There is currently no formal list for DEC-B and an incomplete list of DEC A. However, EOPS and SAMGs have been developed to mitigate DECs. On the other hand, the list of initiating events in the PSA is sufficiently comprehensive; therefore, this deviation will not result in a more than minimal increase in PSA baseline risk. Nevertheless, the improvement actions associated with this deviation to develop a complete list of PIEs for DSA and improve safety are included in the IIP and will be prioritised commensurate with its risk.

Table D.2-1: *Impact of Deviations on DiD Levels*

Deviation Nº	Deviation title	Level 1	Level 2	Level 3	Level 4a[6]	Level 4b[7]	Level 5	Assessment Comments
5A-04-D1	Absence of a list of severe accident phenomena (DEC-B) and an incomplete list of complementary accidents (now referred to as DEC-A) and missing sequences considered for DEC analyses as described in the SAR.				X	X		This deviation identified the need to consider external events, internal hazards, containment bypass events and combination of events in developing the list of DEC A and DEC B. Margins to cliff edges need to be considered and sequences challenging containment for DEC A and DEC B should be included in the PIEs. The existing list of complementary accidents and the EERT screening study for external hazards provide a good starting point for a list of DEC-A accidents, however, the selection will need to be verified against local regulations and international standards. For DEC-B, severe accident phenomena considered in SAMGs provide a good starting point for a DEC-B list. Some DEC-A (complementary accidents) but no DEC-B accidents are documented in the SAR, however, the safety evaluation process does explicitly deal with these issues when plant changes are evaluated, which does provide compensation for the lack of documentation in the SAR. Internal complementary and severe accidents are already taken into account in the PSA. If new DEC-A and DEC-B accidents are identified, they will be incorporated into the PSA model. EOPs and SAMG's may be taken into consideration in the strategies utilized to bring the plant to a safe end state. The improvement actions to complete the DEC A and

Table D.2-1: *Impact of Deviations on DiD Levels*

Deviation Nº	Deviation title	Level 1	Level 2	Level 3	Level 4a[6]	Level 4b[7]	Level 5	Assessment Comments
								DEC B analyses to address this deviation will be prioritised commensurate with its risk.
5B-08-D1	Design Extension Condition (DEC) analysis not aligned with IAEA and WENRA with respect to the use of DEC qualified SSC's, applicable analysis methodology and evaluation of cliff edge effects.				X	X	X	To treat this deviation, the DEC studies that have already been done need to be reperformed without taking credit for the control system, non-classified SSCs or SSCs beyond the qualification limits (where this has been the case). The margins to cliff edge effects also need to be evaluated systematically. Although the DEC analyses are not yet available for the safety demonstration, the available provisions such as the EOPs and SAMG's may be taken into consideration in the strategies utilized to bring the plant to a safe end state.
5F-03-D7	Unjustified Safety Injection (SI) delivery curves used for Emergency Operating Procedures (EOPs)			X				This deviation relates to the lack of justification for the SI delivery curve used in the EOP. The KNPS EOPs are based on international experience and guidelines. The difference in the SI delivery curve is a documentation issue and this will not prevent/inhibit the EOP's from fulfilling their mitigation function during an accident. KNPS EOPs have been validated. Plant-specific calculations will be done and documented in the EOP bases and this action is included in the PSR IIP.

Table D.2-1: *Impact of Deviations on DiD Levels*

Deviation Nº	Deviation title	Level 1	Level 2	Level 3	Level 4a[6]	Level 4b[7]	Level 5	Assessment Comments
7A-30-D1	The approved site safety report for Koeberg is currently outdated and does not reflect the latest site related studies.		X	X	X			The Duynefontyn Site Safety Report (DSSR) is currently in the process of being updated. An expedited seismic evaluation process is being adopted while the SSHAC studies and seismic PSA models are being finalised to conduct a probabilistic seismic reassessment of the plant, as agreed with the NNR. Also, see section 9.3.
7B-04-D1	The parameters associated with air-borne missiles mobilised by high winds are not included in the site safety report.			X	X			The Duynefontyn Site Safety Report (DSSR) is currently in the process of being updated where the effects and mitigation associated with air-borne missiles mobilised by high winds will be included, and there are pending EERI modifications specifically to maintain and protect equipment for mitigation of such events. Most safeguard equipment is situated in concrete buildings which would provide significant protection from airborne missiles. The major vulnerability of the plant is due to the loss of electrical supplies specifically, loss of offsite supplies, and the vulnerability of the EDGs (that is, radiators). Geographical separation of unit EDGs provides a certain degree of protection to the leeward side as well as the availability of mobile diesel generators and accident procedures for the loss of off-site power. The update of the DSSR is included in the PSR IIP. Also, see section 9.3.

Table D.2-1: *Impact of Deviations on DiD Levels*

Deviation Nº	Deviation title	Level 1	Level 2	Level 3	Level 4a[6]	Level 4b[7]	Level 5	Assessment Comments
7F-01-D1	Lack of Internal hydrogen explosion risk assessment for the plant (similar to 1B-20-D1)			X	X	X		The Explosion Hazard Report on the risk of an internal hydrogen explosion concluded that "buildings housing safety related equipment required to safely shut down the plant will not experience an overpressure from any source to the point where severe damage is expected." However, some damage to the radiators of one EDG may occur. A hydrogen explosion of this type and magnitude is considered to be of a relatively low frequency due to a low record of incidents. It can be conservatively assumed that this risk would result in 1E-05 contribution to CDF. As further mitigation, there has been significant progress made in the HAZLOC programme management with the implementation of modifications, processes and procedures aligning with best practice requirements. Safety improvements to address this risk is included in the PSR IIP.
7H-02-D1	The External Hazards Design Basis remains based on the original site studies (characterisation) and does not take into	X	X	X	X	X		The Duynefontyn Site Safety Report (DSSR) is currently in the process of being updated. An expedited seismic evaluation process is being adopted while the SSHAC studies and seismic PSA models are being finalised to conduct a probabilistic seismic reassessment of the plant, as agreed with the NNR. Also, see section 9.3.

Table D.2-1: *Impact of Deviations on DiD Levels*

Deviation Nº	Deviation title	Level 1	Level 2	Level 3	Level 4a[6]	Level 4b[7]	Level 5	Assessment Comments
	consideration more recent site related studies.							
7I-03-D1	Modifications previously identified and prioritised to mitigate the effects of external hazards which exceed the design basis at Koeberg remain unimplemented.				X	X	X	Implementation of the mods may improve the plant's capability to prevent and/or mitigate accidents initiated by external events. These safety improvements are included in the PSR IIP.
7I-03-D2	Recommendations made during the EE-SRA have not been implemented nor dispositioned.			X	X	X	X	Implementation of the recommendations may improve the plant's capability to prevent and/or mitigate accidents initiated by external events. These safety improvements are included in the PSR IIP.

D.3 Impact of Deviations on Fundamental Safety Functions

Table D.3-1: *PSR Deviations Affecting Fundamental Safety Functions*

Deviation Nº	Deviation Title	Reactivity Control	Core Cooling	Confinement of Radioactive Material	Assessment Comments
1E-18-D1	Accuracy of the flow indicators for balancing of the injection flow on the high-head injection lines to the cold legs is insufficient	X	X		The balancing of the injection flow is performed during periodic testing. Three clamp-on ultrasonic flow meters are used to compare the injection flow. For the range of flows, the ultrasonic readings were always within 5% of the indication. The requirement of a balanced flow is therefore assured using an alternative methodology. A safety improvement to address this risk is included in the PSR IIP.
1E-19-D1	Both trains of the component cooling system susceptible to common cause failure during a flooding event		X		This plant deficiency is already incorporated into the baseline PSA model. Incident procedures for loss of component cooling do exist and some mitigations have been implemented that is, a more visible sump level red alarm. The core damage frequency due to flooding of the pump rooms to be 5.5E-08. A Modification has been raised to deal with this plant deficiency and included in the PSR IIP.
1E-30-D1	Control Room Envelope (CRE) Confinement Questionable Performance	X	X	X	Inadequate CRE confinement poses a risk to the control room operators because the dose criterion may be exceeded during certain accident conditions. However, a safety justification performed concludes that the control room habitability criterion of 50 mSv is met when taking into account actual measured plant input parameters. Based on the safety justification the initial high risk deviation identified by the PSR is reduced. Nevertheless, several mitigating actions have been successfully implemented, and more are planned to reduce the unfiltered in-leakage into the control room.

Table D.3-1: *PSR Deviations Affecting Fundamental Safety Functions*

Deviation Nº	Deviation Title	Reactivity Control	Core Cooling	Confinement of Radioactive Material	Assessment Comments
					The improvement actions associated with this deviation will be prioritised commensurate with its risk.
7B-04-D1	The parameters associated with air-borne missiles mobilised by high winds are not included in the site safety report.		X		The Duynefontyn Site Safety Report (DSSR) is currently in the process of being updated where the affects and mitigation associated with air-borne missiles mobilised by high winds will be included, and there are pending EERI modifications specifically to maintain and protect equipment for mitigation of such events. Most safeguard equipment is situated in concrete buildings which would provide significant protection from airborne missiles. The major vulnerability of the plant is due to the loss of electrical supplies specifically, loss of offsite supplies, and the vulnerability of the EDGs (that is, radiators). Geographical separation of unit EDGs provides a certain degree of protection to the leeward side as well as the availability of mobile diesel generators and accident procedures for the loss of off-site power. The update of the DSSR is included in the IIP. Also, see section 9.3.
7I-03-D1	Modifications previously identified and prioritised to mitigate the effects of external hazards which exceed the design basis at Koeberg remain unimplemented.		X	X	Implementation of the mods may improve the plant's capability to prevent and/or mitigate accidents initiated by external events. These safety improvements are included in the PSR IIP.

Appendix E
Requirements for the LTO Checklist

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
5.3.1	Safety case and submissions to NNR for Long Term Operation	What are the regulatory requirements, codes and standards related to AM and LTO, are they consistent with the IAEA Safety Standards, and are the gaps, if applicable, addressed by the plant in the LTO programme?	Req.16, 4.53	1.10, 3.2, 7.2		240-134382460 (PSR Basis Document) Refer to SF-4.5 [44] Safety Case for LTO §9.5 (Ageing Management for LTO)
		Are the AM and LTO activities overseen by the regulatory body throughout the lifetime of the nuclear power plant?	Req.16, 4.53	3. 6, 3.18, 7.39, 7.40		Safety Case for LTO §9.5 (Ageing Management for LTO)
		What are the interfaces between regulatory requirements, codes and standards for LTO and PSR?	Req.16, 4.53	7.2		240-134382460 (PSR Basis Document)
		Is there an adequate regulatory process to ensure safe LTO?	Req.16, 4.53	7.8		NNR NIL-01 Var19 (Koeberg Nuclear Installation Licence) 240-134895976 Koeberg Long Term-Operation – Licensing Strategy

[8] Questions are addressed in the documents listed below.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		Does the PSR provide comprehensive information on AM, equipment qualification and LTO (for example, assumptions, activities, evaluations, assessments and results of the plant programme for AM, equipment qualification and LTO)?	Req.12, 4.44, Req.14, 4.50, Req.16, 4.53	4.3, 4.6-4.8, 5.73, 7.37	3.8, 5.29, 5.42-5.44, 5.49-5.51	240-149139512 (Ageing Management Standard) 240-153546869 (PSR Safety Factor 4 Requirements Ageing Management Report) 240-156945472 (SALTO Ageing Management Assessment Report (Interim)) provides the justification for LTO and AM. 240-153546180 (PSR (Safety Factor 3 Requirements for Equipment Qualification Report) EQ Programme (EQ procedures: 331-186 and 331-219) 331-608 (Global Assessment Report and Integrated Implementation Plan)
		Does the PSR consider the entire planned period of long-term operation and not just the ten years until the next PSR? Is the policy, principles and concept for AM and LTO adequately documented in the PSR report?	Req.12, 4.44, Req.16, 4.53	4.3, 5.74, 7.2, 7.7, 7.38	3.7	NNR NIL-01 Var19 (Koeberg Nuclear Installation Licence) 240-134382460 (PSR Basis Document) 331-608 (Global Assessment Report and Integrated Implementation Plan)
		Does the scope of PSR review identify life-limiting features of the plant in order to determine if there is a need to modify, refurbish or replace certain SSCs for the purpose of	Req.12, 4.44, 4.47, Req.16, 4.53	1.7, 7.15, 7.40	3.2, 3.5	240-134382460 (PSR Basis Document) §2.2 331-608(Global Assessment Report and Integrated Implementation Plan)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		extending the operating lifetime of the nuclear power plant?				
		Is the scope of national and international requirements, codes and standards, as well as practices used in the PSR appropriate and identified in the PSR basis document?	Req.12, 4.44	4.6	4.6-4.9	240-134382460 (PSR Basis Document) §2.3 331-608 (Global Assessment Report and Integrated Implementation Plan)
		Does the Periodic Safety Review, aimed at providing justification of the adequacy of AM for the planned period of long-term operation, focus on safety factors 1 - 4 (plant design, the actual condition of SSCs important to safety, equipment qualification, and ageing) and considers adequately safety factors 8, 9, and 10 (safety performance, use of experience from other plants and research findings, and management system that addresses quality management and configuration management)?	Req.12, 4.44, Req.14, 4.50, Req.16, 4.53	4.6, 4.8	3.6, 3.8	240-134382460 (PSR Basis Document) 331-608 (Global Assessment Report and Integrated Implementation Plan)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		Does PSR review to identify trends of reported events and their possible connection with the degradation of SSCs?	Req.12, 4.44	2.7, 3.35, 4.8, 5.56, 7.40	2.5, 5.94, 5.95	240-134382460 (PSR Basis Document) Trends are covered in SF-2, SF-4 and SF-8 (part 1) SALTO and TLAAs deals with the ageing degradation of SSCs. 331-608 (Global Assessment Report and Integrated Implementation Plan)
		Are the results of the previous PSR examined in order to detect any long-term trends in deteriorating safety performance?	Req.12, 4.44, Req.16, 4.53		2.5, 5.94, 5.95	240-134382460 (PSR Basis Document) Previous SRA-II results were assessed during safety factor reviews to establish the status of any previously identified ageing-related issues. 331-608 (Global Assessment Report and Integrated Implementation Plan)
		Is long-term operation properly justified by safety assessment (that includes scope setting, AMR and revalidation of TLAAs), with consideration given to the life-limiting processes and features of SSCs in the scope of the evaluation?	Req.16, 4.53	2.30, 5.61	2.31,	240-134382460 (PSR Basis Document) 240-153546180 (PSR (Safety Factor 3 Requirements for Equipment Qualification Report) SALTO Assessment (Ageing Management review and revalidation of EQ TLAAs) 331-608

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
						(Global Assessment Report and Integrated Implementation Plan)
		Does PSR global assessment provide safety justification for proposed long-term operation by evaluating the cumulative effects of both ageing and obsolescence on safety and reflecting the combined effects of all safety factors (findings and proposed improvements)?	Req.16, 4.53	2.5, 2.30, 2.32	2.17, 4.21, 4.26-27, 6.6-6.9, 6.12, Appendix II.5	240-163876252 (KNPS 3rd PSR Global Assessment and Integrated Implementation Plan Methodology) 331-608 (Global Assessment Report and Integrated Implementation Plan)
		Is the PSR prepared (for example, the development of a "basis document") and conducted in cooperation with the regulatory body? Is the PSR report that demonstrates safety for long-term operation provided to the regulatory body for review and approval at a level of detail, and in a manner adequate for this purpose?	Req.12, 4.45, Req.16, 4.54	7.40	4.5, 4.6, 6.6-6.9	240-134382460 (PSR Basis Document) developed in line with RG-0028 (Periodic Safety Review of Nuclear Power Plants) and NNR NIL-01 Var19 (Koeberg Nuclear Installation Licence) 331-608 (Global Assessment Report and Integrated Implementation Plan)
		Does PSR review determine reasonable and practicable modifications to be made in order to ensure that a high level of safety is maintained during long term operation? Is the justification for any improvements that cannot	Req.12, 4.47, Req.16, 4.54	1.7, 7.15, 7.40	3.5, 3.6, 3.10, 4.26-4.27, 5.12, 6.6-6.9, 8.14	331-608 (Global Assessment Report and Integrated Implementation Plan)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		reasonably and practicably be made provided?				
		Does the integrated implementation plan to be developed after the PSR contain reasonable and practicable safety improvement?	Req.12, 4.47, Req.16, 4.54	1.7, 7.15, 7.40	2.18, 4.25, 6.7, 8.23, 9.1	331-608 (Global Assessment Report and Integrated Implementation Plan)
6.1	Ageing management, general considerations	Does the plant have a process to ensure competent human resources for LTO including external support?	Req.2, 3.4-3.7			240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) [75] Refer to Appendix F.
		Does the plant have an adequate process for assessing and meeting the organizational competency requirements to support LTO?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix F.
		Have all key technical competencies for LTO activities been identified and do all involved staff meet these requirements?	Req.3, 3.8-3.9, Req.4, 3.10-3.11			240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix F.
		Do personnel assigned to LTO duties that can affect safety have a sufficient understanding of the plant and its safety features?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix F.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		Does plant management have the necessary management skills, experience and knowledge needed to manage safe LTO?	Req.5, 4.1-4.3			240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix F.
		Is the opportunity given to managers and plant personnel to learn from external peer organizations and their lessons learned?	Req.24, 5.27			240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix F.
		Does the plant have an appropriate plant recruitment policy for LTO?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix F.
		Does the policy and role of plant management support training needs and allocate sufficient resources?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Implementation of SAT (Systematic Approach to Training Process Implementation)
		Are personnel involved in LTO activities well trained through on-job-training and other appropriate processes?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) KGT-088 (On-Job Training and Task Performance Assessment) KAA-783 (SAT Implementation Phase)
		Does an appropriate Knowledge Management (KM) policy exist?				240-146686589 (Nuclear Engineering Knowledge Management Standard)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		Are KM principles and practices embedded in the integrated management system?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to F.2.11 and F.2.12 240-146686589 (Nuclear Engineering Knowledge Management Standard)
		Is KM a part of the operating organization's long-term strategy?	Req.4, 3.10, 3.11			240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to F.2.13 As stipulated in 32-83 (Nuclear Management Policy) and all the documents derived from these, the long-term strategy caters for knowledge management. 240-146686589 (Nuclear Engineering Knowledge Management Standard)
		Is there clear ownership of KM processes and issues?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Issue Number F-3. 240-146686589 (Nuclear Engineering Knowledge Management Standard)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		Are KM principles and practices embedded in the organization?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Issue Number F-3. 240-146686589 (Nuclear Engineering Knowledge Management Standard)
		Has the plant embedded KM principles and practices in its process for collecting and using operating experience feedback?	Req.24, 5.28, 5.29, 5.30, 5.31, 5.32	2.7, 2.21, 3.3, 3.30, 4.8, 5.8, 7.16, 7.18	5.7, 5.103-110, 8.13, 9.5	240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Issue Number F-3. 240-146686589 (Nuclear Engineering Knowledge Management Standard)
		Has the plant implemented adequate processes for learning from the LTO experiences of other plants?	Req.24, 5.28, 5.29, 5.30, 5.31, 5.32	2.31, 7.16, 7.18	5.103-110	240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Issue Number F-3.
		Does the plant have a process for knowledge-loss risk assessment and mitigation for suppliers, TSOs and outside service providers?		2.26, 2.29, 6.1-6.3		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Issue Number F-3. Implementation of KM in accordance with 240-146686589 (Nuclear Engineering Knowledge Management Standard)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		Does the plant have established adequate processes for transferring knowledge, information and data to/from the vendor, critical equipment/component suppliers, outsourced services and TSOs?		3.4-3.5, 3.10, 3.13-3.14, 3.16-3.18		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Issue Number F-3. 331-23 KGA-035
		Do IT/IS processes support manage information and records and their availability?	Req.31, 8.4			240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) The NOU uses several IT / IS processes and systems to manage its information and records due to its diverse operational nature, as described in its integrated management system.
		Does the plant retain records of traceability, rationale and assumptions of why and how operational, maintenance and design changes (corporate memory) have been made?		4.1-4.2, 4.9-4.10, 4.13-4.14		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Issue Number F-3. 238-6 (Nuclear Document and Records Management Requirements) 238-8 (The Nuclear Safety and Quality Manual)
6.2.1	Management of ageing throughout the lifetime of the facility, design	Does the plant have access to design basis documentation which contains design basis requirements and supporting design information?		4.13	5.25	240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to A.6.4. 240-89284686 (Locating Technical Information in the KOU)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
						238-6 (Nuclear Document and Records Management Requirements)
		Are the design basis information and its changes included in the safety analysis report or in a separate design basis documentation?		4.14		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to A.6.5 Info is contained in the SAR, DSEs, 240-132364298 (Initial list of Koeberg TLAAs) and other documents.
		Are plant programmes and analyses relevant to AM and evaluation for long-term operation properly documented in the safety analysis report (or in other current licensing basis documents)? Does the information clearly and adequately describe the current licensing basis and the design basis requirements for the plant?	Req.1, 3.2(e)	3.11, 4.1 - 4.2	3.9	240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to A.7.1
6.3.1	Safety analysis report and other current licensing basis documents	Is the justification for plant safety during the planned period of LTO properly documented in the safety analysis report (both ageing aspects and safety upgrades)?	Req.1, 3.2(e)	4.3, 4.10	3.9	240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to A7.2
		Is the safety analysis report being updated to reflect the results of AM and LTO	Req.1, 3.2(e)	4.4, 7.36	3.9	240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		assessment activities (for example, AMR, review of AMPs and plant programmes, revalidation of TLAAs)?				Refer to A.4 and A.7
		Does the safety analysis report update include information describing the assumptions, activities and results of the plant programme for long-term operation (including documentation of the revalidation of the TLAAs for the period of long-term operation)	Req.1, 3.2(e)	4.5, 7.36	3.9	240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to A.4 and A.7
6.3.2	Configuration and modification management programmes including design basis documentation	Does the management system contain processes and activities relating to the configuration management programme and the modification management programme?	Req.10, 4.38, Req.11, 4.39 - 4.43	4.12		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer A.6
		Are all modifications to the plant (relating to the plant configuration: SSCs, process software, OLCs, operating procedures, as well as relating to management systems: organizational structures, operation, and safety assessment tools and processes) properly documented and retained in an auditable and retrievable form? Are all safety	Req.10, 4.38, Req.11, 4.39 - 4.43	4.10		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to A.6.2

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		significant modifications addressed in the SAR?				
		Is a design authority properly established including its role within configuration and modification management?	Req.1, 3.2(f)	4.11		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to A.6.3
		Are alternative arrangements in place, which compensate for the lack of complete design basis documentation at the plant, for example, a programme of reconstitution of design basis?		4.15	5.25-5.25	240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to A.6.4
6.3.3-a)	Safety-related programmes, Maintenance programmes	Is it clearly defined for each in-scope SC what maintenance programmes (for example, preventive, predictive and corrective) are applied, which ageing effects they manage, what maintenance/inspection methods are used, maintenance frequency, tasks, documentation, records and their storage (for example a database)?	Req.31, 8.1, 8.4-8.5, Req.16, 4.54	4.19, 4.20		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix B.
		Are the results of the scope setting, AMR, and TLAA revalidations adequately reflected in the existing	Req.16, 4.54	4.20-4.22		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix B.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		preventive and predictive maintenance programmes?				
		Are preventive and predictive maintenance programmes periodically evaluated based on new regulatory requirements, vendors' recommendations, past maintenance history and feedback from related operational experience and research results and findings?	Req.31, 8.3-8.5, Req.16, 4.54	3.3, 3.30, 3.33, 3.35, 4.21-4.22		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix B.
		Has the plant evaluated the existing preventive and predictive maintenance programmes used to manage the ageing of in-scope SCs against the nine attributes of an effective AMP for the intended period of operation (that is, including LTO)?	Req.31, 8.3-8.5, Req.16, 4.54	4.17, 4.21, 4.22		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix B.
		Are the measures taken to ensure that spare parts are stored in an appropriately controlled environment to avoid degradation mechanisms owing to their storage environment (for example, high or low temperatures, moisture, chemical attack, dust accumulation; for mechanical, EI&C, and civil as applicable)?	Req.31, 8.15, 8,17	3.28		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix B.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
6.3.3-b)	Safety-related programmes, Equipment qualification programme	Has the plant developed, implemented, maintained and periodically reviewed a comprehensive equipment qualification programme including its documentation and consistent with the IAEA safety standards?	Req.13, 4.48-49, Req.16, 4.54	4.23-31		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix D.3.2
		Is there an equipment qualification master list containing mechanical, electrical and I&C components in place? Does it include cables, connectors and penetrations? Is this list updated regularly?	Req.13, 4.48	4.29-4.30		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix D.3. A comprehensive list of components subject to equipment qualification requirements is given in the equipment qualification master list (EQML) provided in EQMM 331-219.
		Does the plant use appropriate seismic motions based on the latest knowledge, operational experience and research findings for seismic qualifications? Are possible ageing effects considered for seismic qualification?	Req.13, 4.48	4.30		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix B.
		Are the results of the scope setting, ageing management review, and TLAA revalidations for LTO adequately used to update equipment qualification programmes?	Req.13, 4.48, Req.16, 4.54	4.23, 4.28-4.30		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to A.5 and D.3.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		Is equipment qualification status preserved and updated through surveillance, maintenance, modifications and replacement, environment and equipment condition monitoring and configuration management? Are adequate interfaces with related programmes in place?	Req.13, 4.48	3.35, 4.18, 4.27, 4.30, 4.31		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to D.3 and D.6.
		Has the plant evaluated the existing equipment qualification programmes for in-scope SSCs against the nine attributes of an effective AMP for the intended period of operation (that is including LTO)?	Req.13, 4.48, Req.16, 4.54	4.17		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Provisions are made in line with RG-0027, AM Standard and 331-148.
		If the equipment qualification programme was designed according to earlier standards, is the re-qualification programme for in-scope SCs in place, focused on ensuring that the equipment can perform its function under the current design basis condition?	Req.13, 4.48, Req.16, 4.53	4.28, 4.30		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to D.3.
		Has it been demonstrated that environmental qualification will remain valid over the expected period of LTO? Does the demonstration support the technical justification that ageing	Req.13, 4.48, Req.16, 4.54	4.25, 4.26, 4.28, 4.30, 5.25(6)		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to D.3.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		effects will be managed effectively? Is timely replacement of equipment that cannot be qualified for the planned period of LTO adequately considered? Has a specific programme for the replacement of mechanical, electrical and I&C equipment with qualified or stated lifetimes less than the planned LTO period been developed and implemented?				
		Do the qualification results on safety-related mechanical, electric and I&C equipment located inside containment specify whether the equipment has been qualified to perform its safety functions in environmental conditions equivalent to design basis accident conditions for the planned period of LTO?		4.25, 4.26, 4.28		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to D.3.
		Is equipment qualification status documented and maintained throughout the life of the plant and consistent with the IAEA Safety Standards?	Req.13, 4.49	4.31		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to D.3.
		Were all identified TLAAs revalidated using methods and	Req.16, 4.54	5.66-5.68, 7.14(b), 7.17,		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		criteria consistent with the IAEA recommendations?		7.18(d), 7.28		Refer to D.3.3
		What corrective or compensatory measures are taken in case TLAAs cannot be revalidated?	Req.16, 4.54	3.34, 5.68		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to D.3.3.
		Is the revalidation of TLAAs documented in an update to the Safety Analysis Report?	Req.16, 4.54	5.70-5.72, 7.36		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to D.3.3.
6.3.3-c)	Safety-related programmes, In- service inspection programmes	Does the ISI programme for the in-scope SSCs clearly identify which ageing effects they manage, the inspection method, the links with AM programmes, the frequency, extent and tasks?	Req.31, 8.1, 8.4-8.5, Req.16, 4.54	4.32-4.34, 4.36		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.3.
		Are results of the scope setting, AMR, and TLAA revalidations for LTO adequately reflected in the existing ISI programmes?	Req.16, 4.54	4.32, 4.35		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.3.
		If ISI results indicate notable degradation, are similar locations appropriately determined? Are SSCs in redundant subsystems inspected independently to detect possible differences in their ageing behaviour?	Req.16, 4.54	4.35		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.3.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		Are ISI programmes periodically evaluated based on new regulatory requirements, vendors' recommendations, past ISI results, operating experience, new knowledge and research findings?	Req.31, 8.3-8.5, Req.16, 4.54	3.3, 3.30, 3.33, 3.35, 4.33, 4.35		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.3.
		Has the plant evaluated the existing ISI programmes used to manage ageing of in-scope SCs against the nine attributes of an effective AMP for the intended period of operation (that is, including LTO)?	Req.16, 4.54	4.17		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.3.
		Have the methodology, equipment, and personnel, which are part of the ISI process, been qualified according to regulatory requirements, codes and standards, and IAEA safety standards as applicable?	Req.16, 4.54	4.33		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.3.2
		Are ISI results documented in the well-maintained database?	Req.16, 4.54, Req.31, 8.4	4.34, 4.36		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.3. and B.4.2
6.3.3-d)	Safety-related programmes, Maintenance programmes	Does the surveillance programme for the in-scope SSCs clearly identify the surveillance measures, the links	Req.31, 8.1, 8.4-8.5, Req.16, 4.54	4.37-4.38		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.4.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		with AM programmes, the frequency, tasks, documentation, records and their storage (for example, a database)?				
		Are results of the scope setting, ageing management review, and TLAA revalidations for LTO adequately reflected in the existing surveillance programme?	Req.16, 4.54	4.39-4.40		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.4.
		Is the surveillance programme periodically evaluated based on new regulatory requirements, vendors' recommendations, past surveillance results, operating experience, new knowledge and research findings?	Req.31, 8.3-8.5, Req.16, 4.54	3.3, 3.30, 3.33, 3.35, 4.41-4.42, 5.8		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.4.
		Has the plant evaluated the existing surveillance and monitoring used to manage the ageing of in-scope SCs against the nine attributes of an effective AMP for the intended period of operation (that is, including LTO)?	Req.16, 4.54	4.17		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.4.
		Has the plant implemented supplementary LTO-related surveillance programmes, such as reactor pressure vessel	Req.31, 8.1, Req.16, 4.54	4.42-4.44		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		supplementary surveillance programme, controlled ageing management programmes for cables, surveillance programme of concrete etc.?				Refer to C.2.1.13 (RPV), D.2.1.2 (Cables), etc.
6.3.3-e)	Safety-related programmes, Water chemistry programme	Are results of the scope setting, ageing management review, and TLAA revalidations for LTO adequately reflected in the existing chemistry programme?	Req.16, 4.54, Req.29, 7.13-7.16	4.45, 4.48		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.5.
		Has the plant chemistry programme been reviewed based on regulatory requirements, vendors' recommendations, chemistry-related surveillance results, operating experience, new knowledge and research findings?	Req.16, 4.54, Req.29, 7.14	3.3, 3.22-3.23, 3.30, 3.35, 4.46, 4.47, 5.8		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.5.
		Has the plant evaluated the existing chemistry programme used to manage the ageing of in-scope SCs against the nine attributes of an effective AMP for the intended period of operation (that is including LTO)?	Req.16, 4.54	4.17		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.5.
		Are chemistry staff aware of the implications of chemistry parameters on known aspects which could adversely impact	Req.16, 4.54, Req.29, 7.13	4.48		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.5.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		safety during LTO (such as corrosion, erosion, inter-granular stress corrosion cracking, primary water stress corrosion cracking, etc. of SCs within the scope of LTO)?				
		Does the chemistry programme include diagnostic parameters that provide useful information for determining and preventing the cause of unexpected ageing?	Req.16, 4.54, Req.29, 7.15-7.16	3.22, 4.45, 4.47		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.5. Diagnostic parameters are listed in KNC-001 and KNC-002.
6.3.4	Corrective action programmes	Is there a corrective action programme in place to ensure that conditions adverse to quality, such as ageing-related degradation, are identified and that corrective actions commensurate with the significance of the issue are specified and implemented?	Req.1, 3.2(e)	3.25, 4.49		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.6.
		Does the corrective action programme document occurrences of identified ageing-related degradation (conditions adverse to quality) and the methods used to address the degradation, such as evaluation and acceptance, evaluation and monitoring, repair, or	Req.1, 3.2(f)	3.3, 3.25, 3.30, 4.50		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.6.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		replacement? Is such information taken into account as plant-specific operating experience?				
		Does the corrective action programme document the modifications to AM programmes, system configuration or plant operations that are made to manage the occurrence or the severity of the ageing effect?	Req.1, 3.2(f)	4.51		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.6.
		Are the corrective action programme and the associated plant-specific operating experience routinely reviewed by individuals responsible for the relevant AM programme to determine whether AM programmes need to be enhanced?	Req.1, 3.2(f)	3.3, 3.30, 3.35, 4.52		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.6.
		Are the modifications of the existing AM programmes specified and implemented, or were new AM programmes developed, if it is determined as needed through the evaluation of the corrective action programme and the associated plant-specific operating experience?	Req.1, 3.2(f)	3.25, 3.30, 4.53		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.6.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
6.4.1	Management of Ageing, Organizational arrangements	Are the roles and responsibilities of all organizations that participate in AM and LTO preparation properly defined and coordinated?	Req.3, 3.8-3.9	3.5, 5.4, 5.6		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A. 240-149139512 (The Ageing Management Standard)
		Has the plant adopted a suitable organizational structure for the preparation and implementation of the AM?	Req.14, 4.50	5.1-5.3, 5.5		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A.
		Has the plant adopted a suitable organizational structure for preparation for LTO?	Req.16, 4.53	3.31, 7.3, 7.4		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A.
		Are adequate resources (for example, human resources, financial resources, tools and equipment, and external resources) allocated to support AM and LTO activities?		5.1, 7.4		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A.
		Are personnel involved in AM and LTO activities properly qualified and trained?		5.7, 6.9		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A.
		Do staff involved in AM and LTO activities have specific job descriptions/task responsibilities?		5.4, 5.6, 7.4		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		Do the plant human resources policy and strategy reflect LTO requirements?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A.
		Do management manuals and job descriptions determine roles, responsibilities and delegations of authority for all managers in key positions related to LTO?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A.
		Is good coordination maintained among different plant groups, among the site organizations and contractors involved in LTO?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A.
		Are staffing and resources sufficient to accomplish the tasks assigned?	Req.4, 3.10, 3.11			240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A and F.
		Is the staffing policy directed at retaining a pool of experienced and knowledgeable staff?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A and F.
		Are long-term staffing policy objectives for human resources established and maintained?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A and F.
		Have specific competence requirements for LTO-related positions been identified and are these used in the				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A and F.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		recruitment/selection process for these positions?				
		Is long term succession planning established and implemented?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A and F.
		Do plant managers have the appropriate resources to carry out their assigned LTO responsibilities and accountabilities?				240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix A and F.
6.6	Scope Setting for SSCs	Does the plant have a systematic scope setting process and methodology(ies), documented and applied to all plant SSCs?	Req.16, 4.54	5.14, 5.15		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix B.1.1.
		Are the criteria for SSCs scope setting for AM and LTO consistent with IAEA Safety Standards?	Req.16, 4.54	5.16, 5.17		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix B.1.1.2.
		Were dedicated plant walk-downs used to check the completeness of the list of SSCs whose failure may prevent SSCs important to safety from performing their intended functions in addition to the analysis of plant documentation?	Req.16, 4.54	5.19		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix B.1.1.6

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		Are the results of the scope setting process clearly and well documented (such as a list of SSCs in scope and out of scope, indicating for example information sources, intended function, safety class, other scoping criteria, etc.)? Are boundaries between SSC within the scope and SSC out of the scope clearly defined?	Req.16, 4.54	5.18,5.20 - 5.21, 5.70, 7.18a), 7.29- 7.30, 7.33		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix B.1.1.5
		Are the boundaries for SCs which include interfaces between different areas (mechanical, electrical, I&C and civil structures) like control valves clearly established?	Req.16, 4.54	5.14, 5.18		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix B.1.1.3
		Have SCs commodities groups (groups of components/ structures which have similar functions, similar materials or are in similar environments) been defined and if so, how?	Req.16, 4.54	5.20		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix B.1.1.7
		Was a list or database of the plant SSCs (for example a master list) used as a basis for the scoping? Are the scoping process results provided in a list of SCs in the scope and a list of SCs out of the scope of AM/LTO?	Req.16, 4.54	5.15, 5.17, 5.19, 5.21, 7.18(a), 7.20, 7.33		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix B.1.1

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		If scope setting data is distributed into more than one database, how is data consistency assured?	Req.16, 4.54	5.14-5.15, 7.20, 7.29-7.30, 7.33		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix B.
		Have SCs commodity groups (group of components/structures which have similar functions, similar materials and are in a similar environment) been defined and if so, how?	Req.16, 4.54	5.20, 7.20		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Appendix B.1.1.7
6.7	Ageing Management Review	Is there a systematic process in place to perform AMR that is consistent with the IAEA safety standards?	Req.14, 4.50, Req.16, 4.54	5.22 - 5.26		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix C, D and E. 240-125122792 (Koeberg Safety Aspects of Long-Term Operation (SALTO) Ageing Management Evaluation Process and Revalidation of the Time-Limited Ageing Analyses).
6.7.1	Ageing Management Review, Identification of relevant ageing effects and degradation mechanisms of structures or components	Does the AMR systematically identify and assess all ageing effects and degradation mechanisms that have been experienced or are anticipated based on an understanding of ageing and to evaluate the impact of ageing on the in-scope SSCs' capability to perform their intended functions?	Req.14, 4.50, Req.16, 4.54	3.24, 5.27, 7.21, 7.23-7.25		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix C, D and E.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		Is the comprehensive understanding of ageing effects and degradation mechanisms for SCs based on design data, fabrication data, operation and maintenance histories, acting stressors (including environmental conditions), results of ISI and surveillance, operating experience and results of research and development, results of walk-downs and condition assessments, and results of the evaluation of TLAAs?	Req.14, 4.50, Req.16, 4.54	5.28, 5.69, 7.21, 7.28		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix C, D and E.
		Is knowledge of the characteristics of the ageing effect (for example, necessary conditions under which the effect occurs and rates of degradation), the related degradation mechanisms and their impact on the structure or component's intended function(s) adequately considered in the identification process?	Req.14, 4.50, Req.16, 4.54	5.29, 7.21		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix C, D and E.
6.7.2	Ageing Management Review, Identification of the appropriate	Were appropriate methods to detect, monitor, prevent and mitigate ageing effects and degradation mechanisms	Req.14, 4.51	5.30, 7.22, 7.24		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix C, D and E.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
	programmes for ageing management	specified for each structure or component?				
		Are existing and proposed plant programmes that support LTO consistent with the IAEA recommendations including the nine attributes?	Req.14, 4.51	3.33, 5.31-5.32, 5.38-5.41, 5.43-5.49, 7.18, 7.20, 7.24, 7.26-7.27		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix C, D and E.
		Is there a process in place to ensure that programmes that are not effective are improved, modified, or new programmes are developed?	Req.14, 4.51	5.32, 7.24		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix C, D and E.
6.7.3	Ageing Management Review, Reporting on the ageing management review	Is the approach to the AMR documented and justified in a way that logically demonstrates that the ageing effects will be adequately managed?	Req.14, 4.51, Req.16, 4.54	5.33, 7.32		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix C, D and E.
		Is all information and conclusions regarding the scope of the AMR documented and include the description and justification of the methods used (methodology), list of SCs subject to the AMR and their	Req.14, 4.51, Req.16, 4.54	5.33-5.34, 7.33		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix C, D and E.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		intended functions, and the information sources to accomplish the above?				
		Does the documentation of the AMR results provide the following information: Current performance and condition of individual SCs Identification of the ageing effects and degradation mechanisms requiring management; Understanding of ageing, monitoring of ageing, prevention and mitigation of ageing effects, as well as information on possible changes in the course of LTO; Identification of the specific programmes or activities that will manage the effects of ageing for each structure, component, or commodity grouping in the scope of the AMR and the need for the development of new AMPs; Description of how the programmes and activities will continue to identify and manage the effects of ageing such that the intended function of the SC will be maintained throughout	Req.14, 4.51, Req.16, 4.54	5.33, 5.35-5.36, 5.70, 7.23, 7.29-7.31, 7.34-7.36		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) The ageing management review (AMR) is performed in accordance with procedure 240-125122792 (Koeberg Safety Aspects of Long-Term Operation (SALTO) Ageing Management Evaluation Process and Revalidation of the Time-Limited Ageing Analyses). Refer to Appendix C, D and E for the process followed with regard to AMR results.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		the planned period of operation or LTO; List of substantiating references and source documents; All information and documentation necessary for an effective management of ageing effects is developed and retained in an auditable and retrievable form.				
6.8	Ageing Management Programmes	Are AMPs and other plant programmes that are credited for managing ageing co-ordinated, implemented and periodically reviewed for improvements? Are they consistent with the nine attributes of an effective AMP?	Req.14, 4.50	3.33, 5.37 - 5.38, 5.46, 7.26-7.27		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Ageing management programmes are structured in line with the nine attributes of an effective AM programme as defined in 331-148.
		If the AMP involves inspection by sampling from a specific population of structures or components, does it describe and justify the methods used for selecting the samples to be inspected and the sample size (with respect to the performance of the SCs intended functions throughout its lifetime)?	Req.14, 4.50	5.41		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)
6.8.1	Development of AMPs	Is the development of the AMPs based on the results of the AMR? Do the AMPs developed include provisions to prevent,	Req.14, 4.50	5.45, 5.48		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Issue Number C-2.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		detect, evaluate and mitigate the ageing effects of anticipated degradation mechanisms, based on the findings from the AMR?				
		Are specific actions relating to the detection, monitoring and prevention or mitigation of ageing effects properly specified within each AMP (these may include maintenance, equipment qualification, in-service inspection, testing and surveillance, as well as for controlling operating conditions)?	Req.14, 4.50	5.44		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Detection monitoring and prevention or mitigation is practised through all AMPs.
		Do all AMPs developed to comply with relevant national regulatory requirements, codes and standards and the AM policy of the plant, and consistent with the nine attributes? Is justification provided if some of the attributes are not met?	Req.14, 4.50	5.46		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to A.1
		Are appropriate acceptance criteria for ageing effects, based on the design basis, technical requirements and applicable regulatory requirements, codes and standards established to facilitate timely corrective actions?	Req.14, 4.50	5.47		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to B.2.1.2

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		Is the information on the current status of in-scope SCs collected for subsequent review of the effectiveness of the AMPs? Are performance indicators representing the effectiveness of the AMPs developed along with the development of the AMPs?	Req.14, 4.50	5.49, 5.56		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Issue Number A-2
6.8.2	Implementation of AMPs	Are AMPs implemented in a timely manner to ensure that the intended functions of structures or components continue to be met? Are data required for decisions on AM actions collected as a part of the AMP implementation?	Req.14, 4.50	5.51, 5.53		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix A.3.2
		Are detailed implementation procedures that describe preventive and mitigatory actions, monitoring or inspection and assessment actions, acceptance criteria and corrective actions established and shared among the different units of the nuclear power plant (for example, the operations, maintenance and engineering units) that are responsible for implementing AM programmes?	Req.14, 4.50	5.52		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Issue Number D-3.
6.8.3	Review and improvement of AMPs	Is the effectiveness of AMPs periodically evaluated in the light	Req.14, 4.50	3.35, 5.54,		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		of current knowledge and feedback from the programme? Are performance indicators, such as material condition, failure and degradation trends, newly revealed ageing, etc. established and used?		5.56		Refer to Issue Number C-2.
		How are AMPs incorporated into the management system of the operating organization?	Req.14, 4.50	5.55		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)
		Are data and information newly acquired through the implementation of AMPs shared among responsible units and other internal or external organizations involved in AM? Are these data connected with the existing plant databases, such as the master equipment and component list?	Req.14, 4.50	5.57		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)
		Is an in-depth review of AM performed periodically (for example, as part of PSR, of safety review for LTO, etc.) and does it demonstrate that ageing effects will continue to be identified and effectively managed? Are the results of the in-depth review documented and do they indicate findings and corrective actions as applicable	Req.14, 4.50	5.61		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		(modifications of existing or development of new AMPs)?				
		Does the plant conclude, after reviewing the existing plant programmes and/or AMPs, that the management of ageing is not adequate in some cases? If so, does the plant modify the existing programme or develop a new programme for the purpose of LTO?	Req.16, 4.54	3.33, 3.35, 5.37, 5.54, 5.58, 5.59, 5.60, 5.63		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)
		Provide selected examples of improved or new AMPs detailed documentation for review (examples to be selected by the reviewer). Does the plant review AMPs for consistency with IGALL AMPs and are areas for improvement in AMPs identified and incorporated?	Req.16, 4.54	5.55, 5.59-5.62		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) For example, Refer to A.5.6. Some new AMPs to be developed are: ∗ Environmental condition monitoring programme; ∗ AMP 212 and AMP 215 – Electrical enclosures and switchgear and other active components, not subject to EQ requirements; ∗ AMP 213 and AMP 218 – Whiskers and capacitors with liquid electrolyte; Electronic equipment not subject to EQ requirements ∗ AMP 220 – Lightning protection and grounding grid are not subject to EQ requirements.
6.9	Time-Limited Ageing Analyses (TLAAs)	Has the plant identified all TLAAs?	Req.16, 4.54	3.34, 5.64,		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
				5.65, 7.14(b), 7.18(d)		A comprehensive list of all the KNPS TLAAs was compiled in L1124-GN-LIS-010, 'Comprehensive List of Koeberg TLAAs'.
		Which methods and information sources were used to identify the TLAAs? Is the identification process (methods and information sources) documented?	Req.16, 4.54	5.64, 5.65, 7.14(b), 7.18(d)		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix C, D and E.
		Were all identified TLAAs revalidated using methods and criteria consistent with the IAEA recommendations?	Req.16, 4.54	5.66-5.68, 7.14(b), 7.17, 7.18(d), 7.28		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix C, D and E.
		What corrective or compensatory measures are taken in case TLAAs cannot be revalidated?	Req.16, 4.54	3.34, 5.68		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix C, D and E.
		Is the revalidation of TLAAs documented in an update to the FSAR?	Req.16, 4.54	5.70-5.72, 7.36		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to Appendix C, D and E.
6.10	Documentation of Ageing Management	Are the assumptions, activities, evaluations, assessments and results of the plant programme for AM and/or for LTO including the list of plant's commitments documented in accordance with	Req.16, 4.53	5.70, 7.29-7.31, 7.33-7.35		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) 240-149139512 (Ageing Management Requirements for Koeberg Nuclear Power

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		national regulatory requirements and consistent with the IAEA Safety Standards in an auditable and retrievable form (see details of refs. provided next)?				Station) meets the requirements of RG-0027.
		Does the documentation include a respective methodologies (for example, in the form of plant procedures, such as for scope setting, AMR, AMP review and improvement, TLAAs identification and revalidation, etc.)?	Req.16, 4.53	7.29-7.30, 7.32,		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) 331-275 (Process for the development and control of AM at Koeberg Operating Unit)
		Does the documentation also include demonstration that ageing effects will be managed during the planned operating period?	Req.16, 4.53	7.35		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) 331-275 (Process for the development and control of AM at Koeberg Operating Unit)
		Does the documentation include an update of the safety analysis report reflecting the assumptions, activities and results of the plant programme for AM, and/or for LTO?	Req.16, 4.53	5.71-5.72, 7.36		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to A.7.
		Are the assumptions, activities, evaluations, assessments and results of the plant programme for AM and/or for LTO reflected in the PSR report? Is the entire	Req.16, 4.53	5.73, 7.37-7.38		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) 331-608 (Global Assessment Report and Integrated Implementation Plan)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		planned period of LTO considered?				
6.11	Management of Technological Obsolescence	Has a dedicated plant programme to manage technological obsolescence consistent with the IAEA safety standards been developed and implemented? Does it address all SSCs important to safety and the spare parts required to maintain these SSCs?	Req.10, 4.38, Req.16, 4.54	3.20, 3.27, 6.1, 6.2		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to D.4. 331-146 (TOMP)
		Does the technological obsolescence programme involve the participation of the engineering, maintenance, operations and work planning units, plant senior management and supply chain organizations?	Req.16, 4.54	6.3, 6.9		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to D.4. 331-146 (TOMP)
		Has the technological obsolescence programme been reviewed for consistency with the 9 attributes? Has it been made available to the regulatory body for review?	Req.16, 4.54	4.17, 6.4, 6.5		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Provisions are made for new programmes to align with the nine attributes as stipulated in RG-0027, AM Standard and 331-148.
		Are technological obsolescence programmes periodically reviewed based on new regulatory requirements,	Req.16, 4.54, Req.24	3.3, 3.30, 3.33, 3.35, 6.10, 6.11		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to D.4.

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		vendors' recommendations, operating experience, and new knowledge and research findings?				331-146 (TOMP) in line with IAEA Specific Safety Guide (SSG)-48 and regulatory RG-0027 within the guidance from EPRI and INPO.
		Does the technological obsolescence programme include the three basic steps (identify and prioritize issues, implement solutions) and activities consistent with the IAEA Safety Standards?	Req.16, 4.54	6.6		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) Refer to D.4. 331-146 (TOMP)
6.12	Reporting	Are efficient data collection and record-keeping systems in place so that trend analyses can readily be performed to predict SSC performance?	Req.15, 4.52	3.23, 5.9-5.12		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) The documentation and records management processes are performed in accordance with KSA-011 (The Requirements for Controlled Documents), KAA-500 (The Process for Controlled Documents), and 331-3 (Document and Records Management Work Instruction) provide requirements for record-keeping.
		Do the data collection and record-keeping systems provide all information for AMR?	Req.15, 4.52	3.23, 5.9-5.12		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) The documentation and records management processes are performed in accordance with KSA-011 (The Requirements for Controlled Documents), KAA-500 (The Process for Controlled

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
						Documents), and 331-3 (Document and Records Management Work Instruction) provide requirements for record-keeping.
		Is design documentation, including documentation from suppliers, available?	Req.15, 4.52	3.13-3.19, 5.9-5.12		240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) The documentation and records management processes are performed in accordance with KSA-011 (The Requirements for Controlled Documents), KAA-500 (The Process for Controlled Documents), and 331-3 (Document and Records Management Work Instruction) provide requirements for record-keeping.
7.3	Principles of and Approach to Long-Term Operation	Does a clear policy exist in the area of AM and LTO, consistent with related IAEA Safety Standards?	Req.16, 4.53, 4.54	3.31, 5.1, 7.7, 7.9		240-149139512 (Ageing Management Standard)
		Does the plant have plant-level documentation covering principles and concepts for AM and LTO?	Req.16, 4.53, 4.54	5.1, 7.5, 7.6-7.8, 7.11-7.15		240-149139512 (Ageing Management Standard)
		Is PSR adequately used to support decision-making for LTO?	Req.16, 4.53	7.27	3.7, 3.10	331-608 (Global Assessment Report and Integrated Implementation Plan)
		Is the plant personnel familiar with the LTO, its principles and concept and is it understood?		7.10		331-608 (Global Assessment Report and Integrated Implementation Plan)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
7.4	Development of a Programme for Long Term Operation	Does the plant have an LTO programme, established in line with the plant's principles and strategy for LTO, and consistent with the IAEA Safety Standards?	Req.16, 4.54	2.31, 3.31 - 3.32, 7.7 - 7.9, 7.16-7.19		331-608 (Global Assessment Report and Integrated Implementation Plan)
		Is the LTO programme a set of activities, including evaluations, assessments, maintenance, inspections and testing, aimed at justifying and demonstrating plant safety for the planned period of long-term operation? Does the LTO programme include scope setting, AMR, review of plant programmes and of AMPs, identification and revalidation of TLAAs, and the development of an implementation programme? Is the LTO programme based on national regulatory requirements and does it consider international best practices, operating experience and research findings?	Req.16, 4.54	2.31, 3.3, 3.30, 4.8, 3.31 - 3.35, 7.7 - 7.9, 7.16-7.19		331-608 (Global Assessment Report and Integrated Implementation Plan) 240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022) 331-608 (Global Assessment Report and Integrated Implementation Plan) 240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)
		Is the LTO programme well documented (for example assumptions, activities, evaluations, assessments and results of the evaluation of AMPs and plant programmes)	Req.16, 4.53-4.54	5.70 7.29		331-608 (Global Assessment Report and Integrated Implementation Plan) 240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		and retained in an auditable and retrievable form?				
		Does the LTO programme address the safety improvements (such as modifications, major reconstructions and scheduled replacements) required as well as the related plant commitments and implementation schedule?	Req.16, 4.53-4.54	7.18e), 7.19, 7.41		331-608 (Global Assessment Report and Integrated Implementation Plan) 240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)
		Does the plant have a programme(s) or action plan for the resolution of issues identified during the review of AMPs, EQ and TLAAs?	Req.16, 4.53-4.54	7.18		331-608 (Global Assessment Report and Integrated Implementation Plan) 240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)
		Has an evaluation of the existing NPP programmes and documentation been performed? Are evaluation results used as a basis for developing the foundation for successful LTO and will they remain effective for the planned period of LTO? Will this evaluation determine if modifications and/or new programmes are necessary to ensure that SSCs are available and qualified to perform their	Req.16, 4.53-4.54	7.11-7.15, 7.16-7.18		331-608 (Global Assessment Report and Integrated Implementation Plan) 240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)

RG-0027 Guidance		Question	IAEA SSR-2/2	IAEA SSG-48	IAEA SSG-25 (RG-0028)	Comment[8]
		intended function for the planned period of LTO?				
		Are recommendations and other suggestions arising from different types of reviews incorporated into plant activities?	Req.12, 4.47, Req.16, 4.53-4.54	2.21, 7.18-7.19, 7.31	9.1-9.5	331-608 (Global Assessment Report and Integrated Implementation Plan) 240-164487877 (Koeberg SALTO Advance Information Pack (AIP) 2022)