REPUBLIC OF SLOVENIA
MINISTRY OF THE ENVIRONMENT AND SPATIAL PLANNING
SLOVENIAN NUCLEAR SAFETY ADMINISTRATION

Slovenian Technical Review Report on the Krško NPP Ageing Management Program

Final Report

December 2017

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Executive Summary

Slovenia has prepared a Technical Report within the Topical Peer Review (TPR) on aging management under the Euratom Directive. The Krško NPP has prepared a preliminary report taking into account the WENRA technical specifications. The report was sent by the Krško NPP to the Technical support organization to review it and prepare an independent expert opinion and to the Slovenian nuclear safety administration (SNSA). The SNSA has reviewed the report and added parts related to regulatory oversight and assessment.

Within the small Slovenian nuclear programme there is one operating nuclear power plant (Krško NPP), one research reactor (TRIGA) and one central storage facility for institutional radioactive waste. In addition, there is also a closed remediated uranium mine at Žirovski Vrh with two remediated disposal sites for mining and milling waste at the site. The Krško NPP, which is covered in the TPR report is a two-loop Westinghouse pressurized water reactor, with thermal rated capacity of 2000 MWt and started its commercial operation on January 15, 1983. The TRIGA Research Reactor has power of 250 kW and is not included in this TPR report.

The Krško NPP intends to extend its operation beyond its original design life to 60 years based on established aging management program (AMP), which is one of the prerequisites for lifetime extension. The AMP was reviewed by an international group of experts and after comprehensive regulatory review also approved by the SNSA. The Krško NPP approach to long term operation is in compliance with U.S. NRC regulations, industry practices and Slovenian legislation. The methodology is in general similar to the IAEA standards and guidelines for ageing management. The AMP fully meets the requirements of NUREG-1801 – GALL. The Krško NPP developed and implemented appropriate programs and procedures according to GALL including methods for the identification and monitoring of the effects of ageing and requirements for the implementation of preventive and corrective measures. The Krško NPP monitors and addresses foreign operational experience in the field of ageing and implements appropriate preventive measures. An important and comprehensive preventive measure in the past was e.g. to replace the reactor vessel head.

Ageing management of electrical cables is covered in the Cable Aging Management Program (CAMP). The program is in compliance with GALL, Maintenance Rule, License Renewal Rule and other EPRI and INPO technical reports and guidelines. The activities of the CAMP provide reasonable assurance that the intended functions of electrical cables and connections exposed to the adverse environments will be maintained consistent with the current licensing basis through the period of extended operation. For ageing management of concealed pipework, the Krško NPP uses program, TD-2Z "Buried and underground piping and tanks", which is in accordance with GALL requirements. The Krško NPP concluded the first cycle of concealed pipework aging management inspections and is preparing for the second cycle. The focus of that one will be on the trending of the results from the first cycle, especially on those pipeline sections where indications of corrosion were found.

The Krško NPP has also prepared and properly implemented aging management programs for monitoring, testing and inspection of the ageing processes of the reactor pressure vessel. No indication of eventual propagation of aging related degradation were found for the reactor vessel components. The SNSA also deems it extremely important that the possibility of hydrogen flaking in the base material of reactor pressure vessel was excluded, according to available information on the technological process of manufacturing, the heat treatment and the results of the pre-service inspection.

The Krško NPP concrete shield building ageing is managed under the program TD-2N "Program tehničnih opazovanj gradbenih objektov in konstrukcij" (Civil Structures Technical Observations Progam). Certain internal criteria in the Krško NPP used in the ageing management program for concrete structures are even more stringent than criteria from industry practice.

The Krško NPP aging management program is a living program constantly being improved based on internal and external operating experiences and results of R&D activities in the world. The Krško NPP ageing management program fully complies with the Slovenian regulation. There are also several challenges and areas for improvement in the future. The most developing and interesting area for R&D is electrical cables. Influence of pressure vessel irradiation on the Krško NPP lifetime is also an important area for R&D in the future. Systematic monitoring and addressing foreign operational experience in the field of ageing, as well as current issues or events in other countries are one of the key elements for safe long term operation. Appropriate preventive measures related to ageing management must also be implemented, what the Krško NPP strives for.

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ABBREVIATIONS

AERM	Aging Effects Requiring Management
ACI	American Concrete Institut
AISC	American Institute of Steel Construction
AM	Aging Management
AMR	Aging Management Review
AMP	Aging Management Program
API	American Petroleum Institute
ASME	The American Society of Mechanical Engineers
ATWOS	Anticipated Accidents With-out Scram
BMI	Bottom Mounted Instrumentation
BWR	Boiling Water Reactor
CAMP	Cable Aging Management Program
CAP	Corrective Action Program
CASS	Cast Austenitic Stainless Steel
CC	Component Cooling Water System
CLB	Current Licensing Basis
CFR	Code of Federal Regulation
CRDM	Control Rod Drive Mechanisms
CUF	Cumulative Usage Factor
DG	Diesel Generator System
DMW	Dissimilar Metal Weld
DO	Diesel Oil Storage System
ED	Engineering Department
ENSREG	European Nuclear Safety Regulators Group
EOL	End of Life
EPRI	Electric Power Research Institut
EQ	Environmental Qualification
FAC	Flow Accelerated Corrosion
FP	Fire Protection
FSAR	Final Safety Analysis Report
GALL	Generic Aging Lessons Learned
HELB	High Energy Line Break
IAEA	International Atomic Energy Agency
I&C	Instrument and Control
ICCMS	Inadequate Core Cooling Monitoring System
IEEE	Institute of Electrical and Electronics Engineers
IGALL	International Generic Aging Lessons Learned
IN	Information Notice
INPO	Institute of nuclear power operations
IPE	Integrated Plant Exemination
IPEEE	Integrated Plant External Evant Exemination
LR	License Renewal
LTO	Long Therm Operation
MEB	Metal Enclosed Bus
MECL	Master Equipment Component List

MR	Maintenance Rule
MRP	Materials Reliability Program
MV	Medium Voltage
NACE	National Association of Corrosion Engineers
NDE	Non-destructive Examination
NEI	Nuclear Energy Institut
NEK	Nuklearna elektrarna Krško (NPP Krško)
NIS	Nuclear Instrumentation
NPAR	Nuclear Plant Aging Research
NPP	Nuclear Power Plant
NPS	Nominal Pipe Size
NSR	Non-safety Related
OE	Operating Experience
PI	Performance Indicator
PSR	Periodic Safety Review
PTS	Pressure Temperature Schock
PWR	Pressurized Water Reactor
PWSCC	Pressurized Water Stress Corrosion Cracking
QA	Quality Assurance
QD	Quality and Nuclear Oversight Division
RB	Reactor Building
R&D	Research and Development
RETS	Radiological Effluent Technical Specification
RG	Regulatory Guide
RHWG	Reactor Harmonization Working Group
RIS	Regulatory Issue Summary
RM	Radiation Monitoring
RPV	Reactor Pressure Vessel
SBO	Station Black-out
SCC	Stress Corrosion Cracking
S/C	Structure/Component
SG	Steam Generator
SNSA	Slovenian Nuclear Safety Authority
SR	Safety Related
SRL	Safety Reference Level
SRP	Standard Review Plan
SSC	Systems, Structures and Components
SW	Essential Service Water System
TD	Technical Division
TLAA	Time Limited Aging Analysis
TS	Technical Specification
TSO	Technical Support Organization
USAR	Updated Safety Analysis Report
US NRC	United States Nuclear Regulatory Commission
WENRA	Western European Nuclear Regulator Association

1 General Information

1.1 Nuclear installations identification

Krško NPP (key parameters):
Name; Krsko NPP
Licensee; Nuklerana elektrarna Krško, d. o. o.
Type of reactor; PWR
Power output; 1994 MW
Year of first operation; 1983
Scheduled shutdown year; 2043

TRIGA Research Reactor (key parameters):
Name; TRIGA MARK II Ljubljana
Licensee; Jožef Stefan Institute
Type of reactor; Light Water Reactor
Power output; 250 kW
Year of first operation; 1966

TRIGA Research Reactor is in operation, but is below the limit for mandatory inclusion and therefore not addressed in the national report (NAR).

1.2 Process to develop the national assessment report

The Slovenian Nuclear Safety Administration (SNSA) participated in all steps of the preparation of the WENRA technical specifications and in an early stage informed the Krško NPP about the TPR process. The SNSA submitted the last revision of the TPR specifications to the Krško NPP.

After the start of the TPR process and according to the process of Extraordinary periodic safety review, the SNSA issued a license amendment. With this decision, the SNSA required from the Krško NPP to perform the TPR and to prepare the technical report in accordance with TPR technical specifications.

The SNSA besides all thematic inspections related to ageing management in the previous years, especially regarding buried and underground piping and reactor pressure vessel, carried out extensive inspection covering all thematic areas of the TPR, namely electrical cables, concealed piping, reactor pressure vessels and concrete containment structures.

The Krško NPP prepared a draft report, which has been reviewed by the SNSA and a technical support organization (TSO). In the next phase the Krško NPP has prepared the final report including the expert opinion prepared by the TSO. The SNSA than reviewed the final report and based on the review proposed modifications and additions to the report. The SNSA has also written parts related to administrative control and evaluation of the Ageing Management Program (AMP).

2 Overall aging management program requirements and implementation

2.1 National regulatory framework

Krško Nuclear Power Plant (NEK) is equipped with a two-loop Westinghouse pressurized water reactor, and has a thermal rated capacity of 2000 MWt. Due to the lack of former Yugoslav nuclear legislation, NEK was erected according to the US NRC regulatory framework (10 CFR 50). NEK started its commercial operation on January 15, 1983. It is co-owned by Slovenian and Croatian state owned companies, and operated by Nuklearna Elektrarna Krško (NEK). NEK intends to continue Krško plant operation beyond its original design life to 60 years and is preparing necessary technical documents to justify the plant's life extension during the Aging Management Review (AMR) project. Based on results of the AMR project, NEK established its aging management program (AMP) with overall AMP documented in MD-5 - Aging Management Program, issued in 2010 (effective date: Feb. 1, 2010 [1]). The international group of experts then independently reviewed the NEK AMP at the end of 2010 [2]. Based on this review NEK updated its AMP and trough communication with SNSA a Decree for license extension of 20 years was issued in 2012 [3].

The objective of this national assessment report is to undertake an assessment, and document the adequacy of NEK's aging management programs which are based on the U. S. Nuclear Regulatory Commission (NRC) license renewal requirements for the extension of the current license term of the US nuclear power plants from 40 to 60 years. The use of U. S. Nuclear Regulatory Commission (NRC) license renewal approach was based on the program for the implementation of the first Periodic Safety Review which was approved by the SNSA's decree [4].

Besides the U.S. NRC legislation, overall aging management program considers the Slovenian legislation: "*Ionizing radiation protection and nuclear safety act* (ZVISJV)" [5], and rule "*Rules on operational safety of radiation or nuclear facilities* (JV9*)*" [6]. Since the only Slovenian nuclear power plant is a Westinghouse designed plant, the development of the Slovenian nuclear legislation is influenced by the U.S. NRC regulatory practice and related legislation.

The "*Ionizing radiation protection and nuclear safety act (ZVISJV)*", and amending acts [5], as well as governmental decrees: "*Decree on practices involving radiation (UV1)*" [7], "*Decree on dose limits, radioactive contamination and intervention levels (UV2)*" [8], and "*Decree on safeguarding of nuclear materials (UV6)*" [9] represent the main national nuclear legislation regulating ionizing radiation protection with the aim of reducing the detrimental effects on human health, and reducing to the lowest possible level the radioactive contamination of the living environment due to ionizing radiation resulting from the use of sources of ionizing radiation, while at the same time enabling the development, production and use of radiation sources and performing radiation practices. The ZVISJV is the main national nuclear law, and represents the main reference that is specifically stated in several other chapters of NEK USAR document.

The "*Rules on operational safety of radiation or nuclear facilities (JV9)"* [6] defines the method of application of operational limits and conditions, method and frequency of reporting, aging management, SSC maintenance, testing and inspection, regular and special reporting requirements, Periodic Safety Reviews, assessment and classification of modifications, and requirements for the emergency planning. The regulation is referenced in the USAR chapters *3.1 Conformance with NRC General Design Criteria and Slovenian Codes and Regulations*, and *17.2 Quality Assurance During the Operations Phase [10]*, as well as in other plant documents, such as plant programmatic documents and procedures.

Requirements for aging management in JV9 are written in articles 14 to 16. In article 14 the basic concepts of aging management are provided:

(1) *The facility operator of a radiation or nuclear facility shall assess the SSCs important to safety taking into account relevant aging and wear-out mechanisms and potential age related JV9: Rules on Operational Safety of Radiation or Nuclear Facilities Unofficial translation 18 degradations, and continuously monitor and assess the condition of SSCs, through their maintenance, testing and inspection.*

(2) *The facility operator of a radiation or nuclear facility shall implement measures to timely detect the inception of aging effects and to allow for preventive and remedial actions. The facility operator shall ensure that the requirements for the achievement of SSC safety functions throughout the service life of the facility are stated in the design bases.*

Article 15 provides requirements for aging management program:

(1) *The facility operator of a radiation or nuclear facility shall have an aging management program to identify all aging mechanisms relevant for structures, systems and components (SSCs) important to safety, determine the possible consequences of aging, and determine necessary activities in order to maintain the operability and reliability of these SSCs. The aging management program shall include, as a minimum:*

1. the criteria for the screening of SSCs to be included in the aging management program;
2. the selection of preventive activities to eliminate or mitigate the effects of aging;
3. the monitoring of modifications of monitored parameters over extended periods of time to establish the time trends of SSC aging;
4. the acceptance criteria for monitored effects of aging;
5. the selection of corrective measures for those SSCs that do not meet acceptance criteria;
6. the organizational aging management; and
7. the instructions for the evaluation of in-house and international experience in the field of aging.

(2) *The aging management program shall consider, as a minimum:*

1. environmental conditions;
2. process conditions;
3. duty cycles;
4. the maintenance, testing and inspection program; and
5. the envisaged operating lifetime of the facility.

(3) *In the case of a nuclear power plant, the aging management program shall include the aging management for mechanical, electrical and civil structure SSCs. In the case of a nuclear power plant, such program shall include the monitoring of the entire primary system pressure boundary, in the case of a research reactor, at least the monitoring of the reactor pressure vessel, if installed, with associated welds. As a minimum, the embrittlement of materials due to effects of neutron flux and of the material fatigue processes due to thermal and other stresses shall be monitored. The measured results shall be compared with predicted levels throughout the facility operating lifetime.*

(4) *The facility operator of a radiation or nuclear facility shall review and update the aging management program in regular time intervals not exceeding the intervals between periodic Safety Reviews, to incorporate new information and knowledge as it becomes available and to use new methods as they become accessible and to assess the performance of the SSC maintenance, testing and inspection program over the facility operating lifetime. Any possible modifications of the program shall be implemented in accordance with articles 34, 35 and 36 of these Rules.*

Article 16 is about reporting on the aging management:

The facility operator of a radiation or nuclear facility shall submit to the Administration the aging management program and any modification of the program or amendment to the program.

As the Slovenian legislation is an integral part of the nuclear power plant operation and practices, and as such the national regulations are strictly followed by the plant staff, and controlled by the regulatory body inspections.

The idea of license renewal was started from the background discussions of the U.S. Atomic Energy Act (Aug. 30, 1954) where it is stated: "*Each such license shall be issued for a specified period, as determined by the Commission, depending on the type of activities to be licensed, but not exceeding 40 years, and may be renewed upon the expiration of such period.*"

In early 1980's, the NRC started looking into the potential concerns for license renewal, and initiated a research program entitled "*Nuclear Plant Aging Research (NPAR)*". The research program resulted in a total of 153 NUREG/CR reports developed and published. These reports described the findings of aging degradation mechanisms and their effects on the most important active and passive systems, structures and components (SSCs) at U.S. nuclear power plants. On the basis of these research studies and the feedback from the U.S. nuclear power industry, the NRC promulgated the License renewal (LR) rule, 10 CFR Part 54 in December 1991.

This Rule was based on two fundamental principles. The first principle is that the regulatory process is adequate to maintain safe operation of the plants except for the detrimental effects of aging. The second principle is that the current licensing basis (CLB) must be maintained and carried forward to the period of extended operation in the same manner and to the same extent. This Rule included both active and passive SSCs, and required licensees to address aging degradation mechanisms that were unique to license renewal. However, a utility demonstration project proved that the intent of the Rule was difficult to implement and was unnecessarily broad.

The NRC subsequently amended the LR Rule and published the Amended LR Rule in 1995. The Amended Rule narrowed the scope to focus only on the passive, long-lived SSCs. The active SSCs with the exception of electrical components were excluded from the scope of license renewal due of self-revealing nature of the aging effects on active components, and the implementation of the Maintenance Rule (MR), i.e. 10 CFR 50.65 code [11] that became effective in 1996. It was judged that the MR should provide adequate maintenance of active SSCs.

The Amended LR Rule [12] has three major technical requirements: the first requirement (10 CFR 54.4) is to identify the SSCs within the scope of license renewal; the second requirement (10 CFR 54.21 (a)) is to perform an integrated plant assessment; and the third requirement (10 CFR 54.21 (c)) is to justify the originally performed time-limited aging analyses (TLAAs) for 60 years of operation.

To simplify and to standardize the review effort, the NRC staff has established a comprehensive license renewal process aiming systematically at scoping and screening all plant SSCs for inclusion in the scope of license renewal, and provide maximum credits for the existing plant programs with no or little duplicate efforts between the existing plant programs and aging management programs for license renewal. This process is documented in the staff document "*Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (SRP-LR)*", NUREG-1800 [13].

The NRC staff also analyzed available information from various sources, including the previous NPAR program findings, the U.S. licensees' event reports related to aging, and 10 industry reports on major component aging (e.g. reactor vessel, RCS piping) which were submitted to the NRC in early 1990's, and developed a regulatory guidance document, NUREG-1801 entitled "*Generic Aging Lessons Learned (GALL) Report*" for the use in industry and staff in preparing and reviewing license renewal applications, respectively.

The development of the GALL report involved more than 100 subject experts from the NRC staff and members of two national laboratories, and the nuclear industry to: 1) establish standard acceptance criteria for an aging management program and 2) evaluate common existing plant programs and to determine whether certain plant programs met the stringent acceptance criteria. If not, the needed modifications to the existing plant programs were recommended. As a result, the GALL report is a compilation of acceptable aging management programs and corrective measures that would be needed for the existing plant programs to become acceptable for S/Cs with specified materials, environments and aging effects.

The U.S. nuclear industry also developed NEI Guideline 95-10 - "*Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule*" to provide further guidance to the licensees for implementing the Amended Rule and for use of the GALL report in the preparation of license renewal applications.

The NRC published Regulatory Guide (RG) 1.188 - "*Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses*" to endorse the NEI 95-10 guideline.

All these guidance documents have since then been updated to incorporate the lessons learned from past reviews and updated documents (including SRP-LR [13], GALL [14], and RG 1.188 [15]), and published as Revision 1 in September 2005. The NEI 95-10 was also updated and were published in June 2005 as Revision 6 [16].

As NPP Krško Aging Management Review Project finished in 2009, Revision 1 of SRP-LR [13], GALL [14], RG 1.188 [15] and Revision 6 of NEI 95-10 [16] were applied.

In 2010, Revision 2 of SRP-LR (NUREG-1800 [17]) and GALL (NUREG-1801 [18]) were issued. Revised SRP-LR and GALL Report issued on 12/16/2010 updated all Aging Management Programs (AMPs) to reflect:

- U.S. and foreign operating experience,
- precedents from License Renewal Applications and staff Safety Evaluation Reports, and
- revisions of industry codes and standards.

Within Revision 2, new GALL AMPs are established for:

- XI.M11B, "Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components (PWRs only)".
- XI.M16A on guidelines in EPRI/Materials Reliability Program reports -227 and -228 for PWR internals inspection and evaluation.
- XI.M40, "Monitoring of Neutron-Absorbing Materials Other than Boraflex", based on ISG.
- XI.M41 for buried and underground piping and tanks, based on material of construction and protection of these S/Cs (such as backfill, coating, and cathodic protection) and the preventive, mitigate and inspection measures to address their aging degradation.

In order to address the changes of regulatory requirements introduced within the revision 2 of NUREG-1801 [18], NPP Krško evaluated the plant's AMP against this new revision. All necessary updates were implemented (NEK ESD-TR-05/14 - "NEK Aging Management Program Evaluation Report") [19].

U.S. NRC 10 CFR 54 approach takes into account the treatment of aging of active components in a separate program, Maintenance Rule (MR). The Maintenance Rule (MR) is required by 10 CFR 50.65 [11], "Requirements for monitoring the effectiveness of maintenance at nuclear power plants". Plants shall monitor the performance or condition of structures, systems or components against licensee-established goals in a sufficient manner to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and where practical, take into account industrywide operating experience. When the performance or condition of a structure, system or component does not meet established goals, an appropriate corrective action shall be taken. The Regulatory Guide (RG) 1.160 "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" [20], and NUMARC 93-01 "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" [21] are used as guidance documents for implementation at NEK.

2.2 International standards

During the past decade, IAEA Member States showed considerable interest in continuing the operation of their nuclear power plants beyond the time originally anticipated (typically, 30-40 years). The IAEA initiated an extra-budgetary program on safety aspects of long term operation of water-moderated reactors in 2003, and published its final report in July 2007 (IAEA-EBP-SALTO). The associated *Safety Series No.57,* on "*Safe Long Term Operation of Nuclear Power Plants*" was published in 2008 [22]. In addition, the IAEA published "*Safety Guide on Aging Management" - NS-G-2.12* [23]. These IAEA documents are essentially consistent with the above U.S. NRC guidance documents for license renewal.

NEK aging management methodology for the preparation for LTO and its license renewal activities are compliant with U.S. NRC regulations (10 CFR 54) and industry practices which are in general, similar to the IAEA standards and guidelines on aging management.

The main difference between the NEK aging management approach and the relevant IAEA guidelines is in the scope/coverage. The scope/coverage of NEK's AMP is in line with the U.S. License Renewal Rule focused on passive and long lived components important to safety. The IAEA screening process includes all SSCs important to safety including active and replaceable components. For NEK (similar to U.S. NRC) approach it can be assumed that aging management of the active components important to safety would be provided for by implementing the U.S. NRC Maintenance Rule (MR) [11]. Nevertheless, NEK's AMP generally follows the IAEA guidance related to programmatic aspects such as AMP organization and methodology.

The IAEA Safety Reports Series No.82 (2015) - "Aging Management for Nuclear Power Plants: International Generic Aging Lessons Learned (IGALL)" [24] compiles information collected from the member states (U.S. NRC provided the IAEA with the Generic Aging Lessons Learned Report [18] Rev 2. - GALL) that has been reflected in the IGALL report), and provides IGALL AMR line items (i.e. only one line for each combination of system, structure/component, aging effect/degradation mechanism, material, environment, AMP, etc. is provided in the IGALL AMR tables). It addresses aging management of passive and active structures and components for water moderated reactors that can have an impact, directly or indirectly, on the safe operation of the plant and that are susceptible to aging degradation. This approach makes the IGALL broader then the GALL, i.e. the IGALL being applicable to a range of nuclear reactors worldwide. The IGALL report was issued in 2015.

The IGALL itself represents an international approach to meet the AM requirements nearly equivalent to the requirements defined in U.S. NRC 10 CFR 50.65 and 10 CFR 54 codes. The GALL approach meets the requirements from U.S. NRC 10 CFR 54 perspective, and is limited to U.S. NRC licensing practice and regulatory concepts and as such is primarily implemented in USA nuclear power plants. U.S. NRC approach takes into account the separate program Maintenance Rule, which meets the requirements from U.S. NRC 10 CFR 50.65 code requirements.

General differences between IGALL and GALL:

- Passive and active safety related SSCs;
- Covering all water moderated reactor designs –PWR (incl. WWER), BWR, CANDU, PHWR;
- AMP – nine IAEA *Attributes* versus ten *Elements* in GALL used to describe AMP.

During May 15 and June 1, 2017, the Operating Safety Review Team (OSART) mission took place in NEK. A part of the agenda was also to review how the IAEA approach of the Long-Term Operation were implemented in NEK. Due to the differences between both approaches the OSART technical notes contained the following findings:

1. In the Aging Management, project walk downs have not been performed systematically, to identify Non-Safety Components Affecting Safety (NSAS), to be included in the scope.
2. Evaluation of active components was not included in the Aging Management project, except for those active components that are Environmentally Qualified (EQ) components, and as such included in the Time Limited Aging Analyses (TLAA) EQ revalidation.
3. The plant has not used risk-based information to extend the scope for Aging Management Review (AMR) in the Aging Management project.
4. Not all parts of the grounding system are included in the scope for AMR in the Aging Management project.
5. Lightning protection Structures and Components (SCs) are not identified in the scope for AMR in the Aging Management project.
6. For civil commodities and some of the electrical, included SCs are not identified by unique identification. The identification is based on physical location or specific function.

7. All TLAAs listed in International Generic Aging Lessons Learned (IGALL) are not considered when establishing the scope of TLAAs to be revalidated.
8. Not all degradation effects documented in IGALL AMP204, Metal Enclosed Bus are considered in AMR AMP-E-05, Bus Duct Commodity.

Within the OSART action plan NEK has already started with the technical resolution of the AMP findings. The plan is as follows:

Ad.1. It is true that no specific walk downs were performed during the aging management review project because the scope was defined based on 10 CFR 54 where also Non-Nuclear Safety systems safety functions influence on Nuclear Safety systems were considered. Additional analysis is recommended by which a review of the seismic walkdowns from the 90's will be performed to identify potential influence of NNSR over the NSR systems. Corrective action is opened.

Ad.2. NEK Aging Management Program is based on US NRC approach and it contains only passive long-lived components/structures. Active components are managed under the Preventive Maintenance Program (PMP), which is in accordance with the requirements of 10 CFR 50.65, "*Requirements for monitoring the effectiveness of maintenance at nuclear power plants"*. No further action is required.

Ad.3. Within the aging management review project NEK did not use probabilistic risk-based approach as NEK used an US NRC approach which did not require it. The finding description contains no requirement. No further action is required.

Ad.4. NEK will study and rationalize supplement NEK AMP electrical components list with additional grounding components. Corrective action is open.

Ad.5. NEK will study and rationalize supplement NEK AMP electrical components list with additional lightening protection components. Corrective action is open.

Ad.6. Designations of mechanical systems components are well ordered within the Master equipment and component list (MECL) database. The commodity groups components or structures are not within the MECL database and did not have the MECL designation as the commodity group. Some of them are additionally included within the MECL database and for some of them this is not feasible. NEK position is that the scope and quality of the AMP is not endangered by such specific approach. No further action is required.

Ad.7. NEK selected all TLAA analyses according to 10 CFR 54 and it may not comply with IGALL methodology. TLAAs are analyses where assumptions of 40 years or effective full power years (EFPY) or similar are taken into account in the initial plant design. NEK will review the whole set of TLAA analyses and will compare them with the requirements from IGALL. If a gap is identified, then NEK will supplement the existing TLAA analyses as required. Corrective action is opened.

Ad.8. NEK will review its AMP for Metal Enclosed Bus and supplement it to fulfil requirements. Corrective action is opened.

As NEK was designed and commissioned according to U.S. NRC regulations, the only rational decision for the development of the aging management program was to choose the U.S. NRC License Renewal Approach in compliance with the U.S. NRC 10 CFR 54 rule.

Following the Fukushima 2011 accident, WENRA mandated its Reactor Harmonization Working Group (RHWG) to review and revise safety reference levels (SRLs) for the existing reactors. The aim of the revision was to integrate the lessons learned from the accident to prevent or control future accidents from similar causes. By revising the SRLs the RHWG took into consideration the IAEA's work to revise its safety requirements, the conclusions of the Second Extraordinary meeting of the Convention on Nuclear Safety, ENSREG recommendations and suggestions as well as national requirements in WENRA member countries. The publication of the revised SRLs reaffirms two of WENRA's main objectives: a harmonized approach to nuclear safety in Europe and the introduction of continuous improvement of reactor safety into the national regulatory framework. In 2015, NEK issued a report NEK-ESD-TR-23/15 - "*NEK Compliance with WENRA Safety Reference Levels*" [25], which evaluated NEK's compliance with WENRA Safety Reference Levels, issued on September 24, 2014 [26].

All chapters of WENRA SRL were reviewed and NEK status is presented in that report. It supports the Plant Safety Upgrade Program and identifies plant areas that need safety improvement in order to satisfy the requirements of WENRA SRL.

The present status report is prepared in order to support the Plant Safety Upgrade Program and to identify the plant areas that need safety improvement, in order to satisfy the requirements of WENRA SRL.

For aging management, WENRA issue I, objectives from I1.1 to I3.2 are relevant. The NEK-ESD-TR-23/15 describes NEK position on the objectives of issues I. No non-compliances were identified for the WENRA Issue I.

2.3 Description of the overall aging management program

2.3.1 Scope of the overall AMP

NEK established its overall AMP at the beginning of 2010 with the issuance of MD-5 "*Aging Management Program*", Revision 0 (Effective date: Feb. 1, 2010) [1]. The NEK AMP is based on U.S. NRC 10 CFR 54, "*Requirements for Renewal of Operating Licenses for Nuclear Power Plants*" [12]. According to NUREG-1801, Rev. 1 [14], and guideline NEI-95-10, Rev.6 [16] that was endorsed by Regulatory Guide 1.188, Revision 1 [15], all important aging aspects for SSCs important to safety including relevant aging mechanisms and required programs were defined.

The scope of the AMP is determined by the 10 CFR 54 process rule [12]. It addresses the aging management of passive, long-lived components, i.e. those that are not subject to routine maintenance or replacement. According to the rule 10 CFR 54.4 (b) the intended functions of the systems, structures and components (SSC) are the bases for including those within the scope of AMP.

The intended functions (as per [16]) define the plant process, condition or action that must be accomplished by SSC to perform or support a safety function for responding to a design basis event or to perform or support a specific requirement of one of the five "*regulated events*" defined in §54.4(a)(3). At the system level, the intended functions shall be the functions of the system that are the reason for including this system within the scope of a license renewal, as specified in §54.4(a)(1)-(3). Where the plant's licensing basis includes specific requirements for system's intended function (e.g. for redundancy, diversity, etc.), then those need to be maintained functional in full also for the period of extended operation. The term »support« here includes system, structure and components whose failure could prevent other SSCs from performing their intended function.

Initially NEK used NUREG-1801, Revision 1 [14], as a collection of evaluated operating experiences and the Generic Aging Lessons Learned. Reflecting the results of dedicated R&D and the new experiences on aging accumulated by that time, in 2010 the NRC issued NUREG-1801, Revision 2 [18], to which NEK AMP is compliant. NEK's report ESD-TR-05/14, Revision 1, "*Aging Management Program Evaluation Report*" [19], documents the evaluation of all 36 AMPs (which encompass all programs to maintain operability and reliability of all SSC subject to aging management) against NUREG-1801, Revision 2. All findings of this evaluation were addressed in NEK's Corrective Action Program. Since then, all NEK AMPs are consistent with NUREG-1801, Revision 2.

Aging of active components at NEK is managed according to U.S. NRC 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" [11] (Maintenance Rule). NEK's program TD-0D [27] assures that the intent of 10 CFR 50.65 is implemented at NEK by using Regulatory Guide (RG) 1.160, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants [20] and

NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants [21] guidelines.

Maintenance Rule in NEK is part of a much wider Equipment Reliability Program. Administrative procedures for Maintenance Rule at NEK are the following:

- ADP-1.1.251, "Obseg programa nadzora učinkovitosti vzdrževanja, kriteriji in uporaba" (Scope of Maintenance Effectiveness Control Program, Criteria and Implementation) [28]
- ADP-1.1.252, "Poročila o stanju sistema in kazalci učinkovitosti" (System Health Reports and Performance effectiveness indicators reports) [29]
- ADP-1.1.253, "Nadzor zanesljivosti opreme" (Equipment Reliability Program) [30]

Preventive maintenance program (PM) has an important role in ensuring a good condition of plant equipment. It is important to perform proper preventive activities within proper time intervals, especially on the most significant plant equipment.

Performance monitoring of systems, structures and components (SSC) condition is essential to detect early indicators of SSC degradation and to act in a timely manner to prevent failures. System Health Reports reflect SSC condition by grading the systems due to their performance, recognize insufficiencies and propose measures for improvement (maintenance activities, equipment replacement, design modifications, etc.).

Long-term planning of investments on the plant is needed to address all necessary equipment replacements or design modifications and to ensure necessary funds for the investments. Proper prioritization of investments gives the proposed time intervals in which the investments are going to be implemented.

a) Assignment of responsibilities within the NEK to ensure an overall AMP is developed and implemented

Based on the resolution of findings of the NEK Aging Management Review (AMR) Project [31], the NEK overall "*Aging Management Program (MD-5)*" [1] document was prepared in 2010 (effective date Feb. 1). The MD-5 is an umbrella AMP plant programmatic document that defines organizational arrangements for the AM related activities. This document specifies responsibilities and actions to ensure that AM programs are implemented, regularly updated, and that all configuration control for AMP is maintained and properly managed.

Responsibilities for the implementation of the overall aging management program are divided among the Engineering Department (ED), Technical Division (TD), Quality and Nuclear Oversight Division (QD) and Training Department. The Technical Director has an overall responsibility for development and maintenance of programs and procedures related to AMP. He is also responsible for keeping records (database) of program results as well as for the implementation of proposed corrective actions in the Aging Management program resulting from AMP experience in the activities on the SSCs in the field.

The main responsibility for the implementation of activities related with the AMP is with the Maintenance Manager. The Maintenance Manager is the dedicated AMP manager during the living phase of the AMP program. The reason why the implementation responsibilities have been assigned to the Maintenance Manager is because the AMP programs are tightly integrated with other plant support activities that are performed by Maintenance department. In effect, he performs the function of the coordinator, similar as it is defined in the IAEA document [22].

The responsibility for the development of AMP Review Summary report every 5 years as defined in MD-5 plant program, is on Licensing Superintendent.

The responsibilities for the Maintenance Rule are on Technical Division (TD).

b) Methods used for identifying SSCs within the scope of overall AMP

The scoping procedures address all 3 criteria specified in 10 CFR 54.4(a) (safety systems, non-safety systems and regulated events). Definitions of the intended functions are consistent with 10 CFR 54.4(b).

Scoping Criteria of the License Renewal Rule [12]:

10 CFR 54.4 Scope

(a) Plant systems, structures, and components within the scope of this part are -

> *(1) Safety-related systems, structures, and components which are those relied upon to remain functional during and following design-basis events (as defined as in 10 CFR 50.49 (b)(1)) to ensure the following functions -*
>
> > *(i) The integrity of the reactor coolant pressure boundary;*
> >
> > *(ii) The capability to shut down the reactor and maintain it in a safe shutdown condition; or*
> >
> > *(iii) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in H 50.34(a)(1), 50.67(b)(2), or § 100.11 of this chapter, as applicable.*
>
> *(2) All non-safety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1)(i), (ii), or (iii) of this section.*
>
> *(3) All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).*

(b) The intended functions that these systems, structures, and components must be shown to fulfill in § 54.21 are those functions that are the bases for including them within the scope of license renewal as specified in paragraphs (a)(1) - (3) of this section.

Plant information to be considered in scoping was specifically identified for each area (mechanical, electrical with I&C and civil structures). The scoping process for structures relies on the results of mechanical and electrical scoping. The civil structures are in scope if they house or they have ability to endanger SSCs important to safety.

For the identification of safety related SSCs, the reference was NEK USAR [10]. Some safety class 3 systems or subsystems were determined to be "out of scope" based on 10 CFR 54.4 (a)(1). These were further evaluated against scoping criteria 10 CFR 54.4 (a)(2) and (3).

For the identification of non-safety related SSCs, the main references used included the NEK USAR, Individual Plant Examination (IPE) and Individual Plant External Event Examination (IPEEE) documents [32]. The non-safety related SSCs falling within the scope were those relevant (relied on) for the following events: HELB, flooding, missiles, heavy load drop, seismic interactions and interactions between support systems and interfacing systems.

To screen the SSCs that are subject to AMP, NEK used components-specific evaluation for mechanical components and commodity-group evaluation method approach for electrical and civil components and structures. Mechanical components were directly taken from the Master Equipment Component List (MECL database) [33]. The evaluation was performed considering the functions of those components within 38 systems (those that were determined to be within the scope of AMP). Not all electrical components were found in MECL database. For the cables, the PCCKS data base provided necessary information (see 3.1). The electrical components were considered as belonging to one of 9 commodities, and civil components/structures as belonging to one of the Safety related buildings or structures. Civil structures are in scope if they house or they have ability to endanger SSCs important to safety.

The methodology for determining which components should be addressed by the aging management program is based on the criteria stipulated by the license renewal rule [12], as quoted below.

Screening Criteria of the 10 CFR 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants [12]:

10 CFR 54.21(a)(1)(i) and (ii)
(1) For those systems, structures, and components within the scope of this part, as delineated in §54.4, identify and list those structures and components subject to an aging management review. Structures and components subject to an aging management review shall encompass those structures and components: *(i) That perform an intended function, as described in §54.4, without moving parts or without a change in configuration or properties. These structures and components include, but are not limited to, the reactor vessel, the reactor coolant system pressure boundary, steam generators, the pressurizer, piping, pump casings, valve bodies, the core shroud, component supports, pressure retaining boundaries, heat exchangers, ventilation ducts, the containment, the containment liner, electrical and mechanical penetrations, equipment hatches, seismic Category I structures, electrical cables and connections, cable trays, and electrical cabinets, excluding, but not limited to, pumps (except casing), valves (except body), motors, diesel generators, air compressors, snubbers, the control rod drive, ventilation dampers, pressure transmitters, pressure indicators, water level indicators, switchgears, cooling fans, transistors, batteries, breakers, relays, switches, power inverters, circuit boards, battery chargers, and power supplies; and* *(ii) That are not subject to replacement based on a qualified life or specified time period.*

Scoping for active components for MR is based on requirements from U.S. NRC 10 CFR 50.65 [11]:

(1) Safety-related structures, systems and components that are relied upon to remain functional during and following design basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in Sec. 50.34(a)(1), Sec. 50.67(b)(2), or Sec. 100.11 of chapter 10, as applicable.
(2) Non-safety related structures, systems, or components:
(i) That are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures (EOPs); or
(ii) Whose failure could prevent safety-related structures, systems, and components from fulfilling their safety-related function; or
(iii) Whose failure could cause a reactor scram or actuation of a safety-related system.

c) Grouping methods of SSCs in the screening process

All SSCs that remain after applying the scoping & screening criteria described above stay in the scope of the aging management review (AMR). The AMR is then performed on those SSCs reflecting the SSCs type, material and environment using NUREG 1801 tables. The GALL report [18] is referenced as the technical basis document as per NUREG-1800, SRP-LR [17]. It identifies aging management programs (AMPs) that were determined to be appropriate (acceptable) programs to manage the aging effects in the scope of license renewal process, as required by 10 CFR Part 54 [12].

The NEI 95-10 guidelines [16] recommend the use of commodity groups of similar structures or components to enable the disposition of an entire group with a single aging management review. The basis for grouping SSCs is the commonality of characteristics such as similar design, similar materials or construction, similar aging management practices and similar environments. If the environment, in which a structure or a component operates, suggests potential different environmental stressors, then the commodity group determination should also consider service time, operational transients, previous failures and any other conditions that might lead to a different outcome in terms of aging.

The Aging Management Review (AMR) at the system/commodity level encompasses 61 reports, each addressing a specific system or commodity group. This includes 38 mechanical systems reports, 9 electrical commodities reports, and 14 civil/structural commodities reports.

Mechanical systems (reports AMP-M-01 to AMP-M-38) include:

- Auxiliary Feedwater System
- Compress Air System
- SG Blowdown System
- Floor and Equipment Drain System
- Condensate System
- Containment Testing and Pressurizing System
- Auxiliary Steam Heating System
- Boron Recycle System
- Containment Spray System
- Fire Protection System
- Feedwater Chemical Addition System
- Waste Disposal Solid System
- Component Cooling System
- Demineralized Water System
- Waste Processing Liquid System
- Refueling Water System
- Ventilation Air System
- Chemical and Volume Control System
- Chilled Water System
- Diesel Generator System
- Diesel Oil Storage System
- Feedwater System
- Waste Processing Gas System
- Hydrogen Control and Monitoring System
- Instrument Air System
- In-Core Instrumentation System
- Main Steam System
- Reactor Makeup Water System
- Reactor Coolant System
- Residual Heat Removal System
- Radiation Monitoring System
- Auxiliary Steam System
- Steel Containment System
- Spent Fuel System
- Safety Injection System
- Sampling System
- Essential Service Water System
- Turbine Plant Sampling System

Civil structures (reports AMP-C-01 to AMP-C-14) include:

- Control Building
- Reactor Building and Steel Containment
- Interior Structures
- Intermediate Building
- Auxiliary Building
- Component Cooling Building
- Decontamination Building
- Diesel Generator Building
- Fuel Handling Building

- SW Pump House & Fire Service
- ESW Water Discharge
- Radwaste Storage Facility
- River Dam
- Components Bases & Supports

Electrical Commodity Groups (reports AMP-E-01 to AMP-E-09) include:

- Cables Commodity
- Electrical Interconnections Commodity
- High Voltage Insulators Commodity
- Grounding Commodity
- Bus Duct Commodity
- Box Commodity
- Terminal Block Commodity
- Splice Commodity
- Connectors Commodity

d) Methodology and requirements for evaluation of the existing maintenance practices and developing of aging programs appropriate for the identified significant aging mechanism

The NEI 95-10 [16] envisages that either a single program/activity or a combination of several programs/activities might be aimed at mitigating identified aging mechanisms. Once the approach is selected (i.e., single program/activity, multiple programs/activities), the potential aging management program/activity shall be evaluated against 10 elements of GALL [18]. Hereafter, aging management program(s), aging management activities or collections of aging management programs and activities used to manage an aging effect will be referred to as an AMP.

The assessment of all aging mechanisms/systems or commodities groups against 10 GALL elements identified NEK programs that could be credited for managing the aging effects relevant to the SSCs within the scope of aging management program. It focused on the programs that existed at NEK at the time of the initiation of the AMR. The assessment verified whether GALL's 10 Elements were adequately addressed in programs and as such could be credited for aging management. The consistency with the NUREG-1801, Revision 1 guidance [14] was also evaluated. In the AMP-SUM-01 [34] evaluation the GALL's 10 Elements were addressed for each program. Thirteen NEK's existing programs were found to adequately fulfil the requirements (10 elements). Further 14 programs were found partially in compliance and were, after modification, credited for in the AMP. Eight new programs were developed to fulfil necessary conditions for the AM. NEK's report AMP-SUM-03 [31] summarized the evaluation undertaken in terms of the qualification of the existing and new programs to be credited in the AMP, but also described changes that were necessary in some of the existing programs.

To comply with the latest applicable requirements, after NUREG-1801, Revision 2 [18] became available, the same evaluation was repeated. As a consequence, 29 programs were found to be adequate/compliance, 6 were modified, and 1 new program was developed. The results were documented in the NEK report NEK ESD-TR-05/14 - "*NEK Aging Management Program Evaluation Report*" [19] that effectively replaced the AMP-SUM-01. With this NEK AMP is fully consistent with the requirements of the latest applicable regulatory document (i.e. Revision 2 of NUREG 1801-compliance requirement in Slovenia). The results of the assessment were presented in the following Tables 2-1 to 2-3, which show the existing, modified and new NEK programs, respectively.

Table 2-1: Existing NEK programs according to NUREG-1801, Revision 2

NUREG-1801 Number	**NUREG-1801, Rev. 2 PROGRAM**	**EXISTING NEK Program**
XI.M1	ASME Section XI In-service Inspection Subsections IWB, IWC and IWD	TD-2E/4; In-service Inspection Program - The 4th Inspection Interval
XI.M2	Water Chemistry	ADP-1.6.021; Kemijske specifikacije in kriteriji za korektivno ukrepanje (Chemical specifications and criteria for corrective actions) CAP-6.001; Plan kemijskega in radiokemijskega vzorčevanja (Chemical and radio-chemical sampling plan)
XI.M3	Reactor Head Closure Studs	TD-2E/4; Inservice Inspection Program - The 4th Inspection Interval
XI.M10	Boric Acid Corrosion	TD-2J; Boric Acid Inspection Program
XI.M11B	Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components (PWRs Only)	TD-2S; Program nadzora inconela 600/82/182 (Inconel 600/82/182 Surveillance program) TD-2R; Program nadzora glave reaktorske posode in BMI penetracij (Reactor Head and BMI surveillance program) TD-2E/4; Inservice Inspection Program - The 4th Inspection Interval
XI.M16	PWR Vessel Internals	TD-2O; Program nadzora notranjih delov reaktorske posode in mehanizmov kontrolnih palic (Reactor Internals and Control Rod Drive Mechanism Surveillance Program) TD-2E/4; Inservice Inspection Program - The 4th Inspection Interval
XI.M17	Flow Accelerated Corrosion	QD4; Program Erozije/Korozije (Erossion/Corossion Program)
XI.M18	Bolting Integrity	TD-2E/4; Inservice Inspection Program - The 4th Inspection Interval
XI.M19	Steam Generator Tube Integrity	TD-0H; Program Uparjalnikov (Steam Generator Program)
XI.M20	Open-Cycle Cooling Water System	TD-1Z1; Open-Cycle Cooling Water System
XI.M21	Closed-Cycle Cooling Water System	TD-1Z2; Closed-Cycle Cooling Water System
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	ADP-1.1.141; Ravnanje s težkimi bremeni v NEK (Heavy Load Handling in NEK) ADP-1.1.142; Uporaba dvigal, dvižnih naprav, viličarjev in pomožnih nosilnih sredstev v NEK (Use of Cranes, lifting devices, fork lifts and auxiliary transport devices in NEK) ADP-1.4.160; Program preventivnega vzdrževanja dvigal, opreme za prenos goriva in pomožnih nosilnih sredstev (Preventive Maintenance program for lifting devices, equipment for fuel transfer and auxiliary transport devices)
XI.M24	Compressed Air Monitoring	ADP-1.4.225; Program preventivnega vzdrževanja kompresorjev (Preventive Maintenance Program for Compressors) GMM-4.550; Splošni postopek strojnega vzdrževanja kompresorjev (Compressor Mechanical Maintenance General Procedure)

NUREG-1801 Number	NUREG-1801, Rev. 2 PROGRAM	EXISTING NEK Program
		TD-2E/4; Inservice Inspection Program - The 4th Inspection Interval
XI.M26	Fire Protection	ADP-1.0.500; Program Protipožarne zaščite-požarni red (Fire Protection Program) OSP-3.4.590 "Vizuelni pregled požarnih barijer" (Fire bariers visual Inspection) OSP-3.4.381 "Mesečni preizkus črpalk sistema protipožarne zaščite (FP)" (Fire Protection Pumps' Monthly Test) OSP-3.4.586 "18 mesečni funkcionalni test črpalk protipožarnega sistema (FP)" (Fire Protection Pumps' 18-month functional test)
XI.M27	Fire Water System	QD-5; Program inspekcije protipožarnega sistema (Fire Protection System Inspection Program) ADP-1.0.500; Program Protipožarne zaščite-požarni red (Fire Protection Program)
XI.M29	Aboveground Steel Tanks	TD-2ZZ; Izvajanje pregledov nadzemnih tankov (Above Ground Tanks Surveillance Program)
XI.M30	Fuel Oil Chemistry	OSP-3.4.308; 3-mesečni test kvalitete goriva za diesel generatorja (Diesel Generator 3-month Fuel Oil Quality Test)
XI.M31	Reactor Vessel Surveillance	ED-5; Reactor Vessel Irradiation Surveillance Program
XI.M35	One-Time Inspection of ASME Code Class 1 Small-Bore Piping	TD-2X; One-Time Inspection of ASME Code Class 1 Small-Bore Piping
XI.M37	Flux Thimble Tube Inspection	TD-2R; Program nadzora glave reaktorske posode in BMI penetracij (see XI.M11A) (Reactor Head and BMI surveillance program)
XI.M39	Lubrication Oil Analysis	ADP-1.4.130; Program podmazovanja strojne opreme Nuklearne elektrarne Krško (NEK Mechanical Equipment Lubrication Program)
XI.M40	Monitoring of Neutron-Absorbing Materials Other Than Boraflex	ED-4; Surveillance Program For Borated Stainless Steel Sheets
XI.S1	ASME Section XI, Subsection IWE	TD-2H/2; Containment Inspection Program
XI.S3	ASME Section XI, Subsection IWF	TD-2E/4; Inservice Inspection Program - The 4rd Inspection Interval
XI.S6	Structures Monitoring Program	TD-2L; Program nadzora gradbenih objektov in konstrukcij v NE Krško (Civil Structures Surveillance Program) TD-2N; Program tehničnih opazovanj gradbenih objektov in konstrukcij (Civil Structures Technical Observation Program) TD-2U; Program nadzora nosilnih jeklenih konstrukcij v NE Krško (NEK Steel Structures Surveillance Program)
XI.E1	Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	TD-2D; Cable Aging Management Program ADP-1.4.451; Program preventivnega vzdrževanja nizkonapetostnih elektromotorje (Low-voltage Motors Preventive Maintenance Program)

NUREG-1801 Number	NUREG-1801, Rev. 2 PROGRAM	EXISTING NEK Program
		ADP-1.4.453; Program preventivnega vzdrževanja visokonapetostnih elektromotorjev (High-voltage Motors Preventive Maintenance Program) ADP-1.4.454; Program preventivnega vzdrževanja MOV aktuatorjev (MOV Actuators Preventive Maintenance Program) ADP-1.4.455; Program preventivnega vzdrževanja nizkonapetostnih stabilnih naprav (Low-voltage Stabile Equipment Preventive Maintenance Program) ADP-1.4.456; Program preventivnega vzdrževanja visokonapetostnih stabilnih naprav (High-voltage Stabile Equipment Preventive Maintenance Program) ADP-1.4.457; Program preventivnega vzdrževanja regulatorjev napetosti in električne zaščite (Voltage Regulator and Over-voltage Protection Preventive Maintenance Program)
XI.E2	Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	TD-2D; Cable Aging Management Program ADP-1.4.600; Program nadzornega preverjanja in umerjanja instrumentacije vgrajene v NE Krško (NEK Instrumentation Surveillance and Calibration Program) ADP-1.4.603; Preverjanje, umerjanje in verifikacija instrumentacijske opreme vgrajene v NE Krško (NEK Instrumentation Equipment Surveillance, Calibration and Verification)
XI.E6	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	TD-2D; Cable Aging Management Program ADP-1.4.451; Program preventivnega vzdrževanja nizkonapetostnih elektromotorje (Low-voltage Motors Preventive Maintenance Program) ADP-1.4.453; Program preventivnega vzdrževanja visokonapetostnih elektromotorjev (High-voltage Motors Preventive Maintenance Program) ADP-1.4.454; Program preventivnega vzdrževanja MOV aktuatorjev (MOV Actuators Preventive Maintenance Program) ADP-1.4.455; Program preventivnega vzdrževanja nizkonapetostnih stabilnih naprav (Low-voltage Stabile Equipment Preventive Maintenance Program) ADP-1.4.456; Program preventivnega vzdrževanja visokonapetostnih stabilnih naprav (High-voltage Stabile Equipment Preventive Maintenance Program) ADP-1.4.457; Program preventivnega vzdrževanja regulatorjev napetosti in električne zaščite (Voltage Regulator and Over-voltage Protection Preventive Maintenance Program) ADP-1.4.600; Program nadzornega preverjanja in umerjanja instrumentacije vgrajene v NE Krško (NEK Instrumentation Surveillance and Calibration Program) ADP-1.4.603; Preverjanje, umerjanje in verifikacija instrumentacijske opreme vgrajene v NE Krško (NEK Instrumentation Equipment Surveillance, Calibration and Verification)

Table 2-2: Modified NEK programs due to NUREG-1801, Revision 2

NUREG-1801 Number	NUREG-1801, Rev. 2 PROGRAM	MODIFY NEK Program
XI.M32	One-Time Inspection	TD-2XX; One-Time Inspection
XI.M36	External Surfaces Monitoring Program	TD-2AA; External Surfaces Monitoring Program
XI.M38	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	TD-2W; Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components
XI.M41	Buried and Underground piping and Tanks	TD-2Z; Buried and Underground Piping and Tanks
XI.S4	10 CRF 50, Appendix J	ADP-1.1.235; Containment Leakage Rate Testing Program
XI.E3	Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	TD-2D; Cable Aging Management Program ADP-1.4.453; Program preventivnega vzdrževanja visokonapetostnih elektromotorjev (High-voltage Motors Preventive Maintenance Program) ADP-1.4.456; Program preventivnega vzdrževanja visokonapetostnih stabilnih naprav (High-voltage Stabile Equipment Preventive Maintenance Program) ADP-1.4.457; "Program preventivnega vzdrzevanja regulatorjev napetosti in elektricne zascite" (Voltage Regulator and Over-voltage Protection Preventive Maintenance Program) ADP-1.4.458; Program preventivnega vzdrzevanja elektricnih instalacij in razsvetljave (Electrical Installations and Lighting Preventive Maintenance Program)

Table 2-3: New NEK program due to NUREG-1801, Revision 2

NUREG-1801 Number	NUREG-1801, Rev. 2 PROGRAM	NEW NEK Program
XI.M33	Selective Leaching of Materials	TD-2XX; One-Time Inspection (inclusion of Selective leaching) (New)

e) Quality assurance of the overall AMP in particular

The NEK's Quality Assurance Program documented in *QD-1* - "*Quality Assurance Plan*" [35], fulfils the requirements of *10 CFR 50, Appendix B* - "*Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants*" [36], and of applicable Slovenian regulations [5] and [6], as well as the requirements of the Appendix A.2 of NUREG-1800 [17], which defines specific QA requirements for the AMP. The QA program includes relevant elements including corrective action, confirmation processes and administrative controls, all of which are applicable to the safety-related and non-safety-related SSCs that are subject to aging management.

The AMP is, similar to all other programs and activities at NEK, subject to QA program's requirements, as follows:

Corrective Actions

The Corrective Action Program (CAP) is applied regardless of the safety classification of the structure or component. The corrective actions are to be implemented through the preparation of the corrective action report, in accordance with plant procedures, for actual or potential problems including unexpected plant equipment degradation, damage, failure, malfunction or loss. In the case of the AMP, where the non-conforming conditions are found (i.e., the acceptance criteria are not met), a corrective action report would be prepared in accordance with those procedures, and addressed through the CAP. Typically, the equipment deficiencies are corrected by initiation of a work order, which is then acted upon.

Confirmation Process

The focus of the confirmation process is then to verify effective implementation of corrective actions. The measure of effectiveness of the confirmation process is to correct adverse conditions and preclude repetition of conditions affecting quality. NEK's procedures include provisions for timely evaluation of adverse conditions, and the initiation of corrective actions as required, including root cause analysis and prevention of recurrence, where appropriate. These procedures are provided for tracking, coordinating, monitoring, reviewing, verifying, validating, and approving corrective actions, to ensure effective corrective actions are taken.

The corrective action process is also monitored for potentially adverse trends, which is of specific relevance for aging management at NEK. The corrective action process, which is required within the aging management program, would also uncover any unforeseen degradation or conditions that were not envisaged within the AMP. This would automatically lead to a corrective action at the SSC level, but also on the AMP-program level.

The corrective actions and confirmation processes are applicable and applied to all safety-related and non-safety-related SSCs encompassing all of those subjects to NEK's AMP. In this respect, NEK's AMP is fully compliant with the requirements for the QA as defined by the NUREG-1801, Chapter XI.

Administrative Controls

Administrative controls procedures provide a formal review and approval process on procedures and other forms of administrative control documents as well as guidance on classifying documents into the proper document type.

According to the applicable Slovenian regulation JV9 [6], record keeping requirements are maintained within NEK Document Control Module (DCM) [37], NEK Master Equipment and Component List (EAM-MECL) [33], NEK Corrective action program (CAP) [38], and NEK Work Order system (WOS) [39]. All those requirements are applicable (and fulfilled) for NEK's AMP. Furthermore, NEK's QD-1 requires that the records for AMP are being kept for full (extended) lifetime.

Efficiency of the NEK Overall AMP is monitored using Performance Indicators (PIs). NEK uses the PI extensively, (more than 160 of different PIs are defined and monitored on continuous basis) to monitor different areas of plant's operational performances. Among those there are two AMP related PIs:

- Number of AMP related corrective actions (3-month PI);
- AMP implementation efficiency (Annual PI).

The first one is used for trending the number of corrective actions performed necessary to rectify (newly discovered) aging mechanism by the implementation of an action on a SSC that is part of the NEK's AMP.

The second PI is used to monitor the status of undertaken AMP actions *vs.* planed AMP on a cycle basis. It shows the pace of the progression on the implementation of the AMP tasks, as NEK is approaching the end of its design life and entering to LTO conditions (in 2023).

2.3.2 Aging Assessment

Originally, NEK's Aging Management activities were initiated within first Periodic Safety Review (PSR) in 2001. At that time the objective was to provide appropriate organizational and technical framework for managing aging of SSCs important to safety immediately, but also considering possible LTO/license renewal.

Given that NEK's licensing conditions are per applicable Slovenian regulations closely aligned with those of the U.S. NRC, NEK in agreement with the SNSA decided that the most appropriate approach to the LTO/Aging management will be the one as defined by the U.S. License Renewal Rule, 10 CFR 54 [12], and related regulatory documents (NUREG-1800 and NUREG-1801). Reflecting the guidance in the NEI 95-10, Rev. 6 [16] and in compliance with NUREG-1801, Rev. 1 [14] (and later with [18]), NEK's AMR selected and subsequently assessed all SSCs important to safety. If any potential aging mechanism were found, then relevant AMPs were identified and credited to ensure that necessary safety functions remain available throughout the extended lifetime.

The Aging management review (AMR) for the structures, components or groups of structures and components selected through the screening process was performed in order to determine appropriate SSCs, specific aging management programs (AMPs). The AMR process aimed at ensuring the understanding of aging of NEK's SSCs and to, subsequently, establish appropriate measures (AMPs) to monitor aging processes and conditions and to mitigate any effects of aging. These basic objectives are fully consistent with Slovenian national regulation and also with internationally recognized standards (e.g. IAEA Ref. [23]).

The AMR task within NEK's AMR Project was aimed at evaluating the adequacy of specific (and existing) NEK's programs to be used for monitoring and/or mitigating consequences of aging phenomena. Therefore, the AMR concentrated on verification that the methodology selected for the AM is consistent with the applicable regulatory requirements and criteria provided in this reference documents.

a) How key standards and guidance, as well as key design, manufacturing and operations documents are used to prepare the overall aging management program

The key standards and guidance, as well as key design, manufacturing and operations documents used for NEK overall AMP are:

- US NUCLEAR REGULATORY COMMISSION, "10 CFR 54 – The License Renewal Rule".
- US NUCLEAR REGULATORY COMMISSION, "Standard Format and Content for LRA", NRC Reg. Guide 1.188.
- US NUCLEAR REGULATORY COMMISSION, "Standard Review Plan for License Renewal", NUREG 1800, Rev.1, Sept. 2005.
- US NUCLEAR REGULATORY COMMISSION, "Standard Review Plan for License Renewal", NUREG 1800, Rev.2, Sept. 2010.
- US NUCLEAR REGULATORY COMMISSION, "Generic Aging Lessons Learned (GALL) Report", NUREG 1801, Rev.1, Sept. 2005.
- US NUCLEAR REGULATORY COMMISSION, "Generic Aging Lessons Learned (GALL) Report", NUREG 1801, Rev.2, Sept. 2010.
- NUCLEAR ENERGY INSTITUTE, "Industry Guideline for Implementing the Requirements of 10 CFR 54 – The License Renewal Rule", NEI-95-10, Rev. 6.
- USAR, "Updated Safety Analysis Report", Nuclear Power Plant Krško"
- Technical Specifications, Nuclear Power Plant Krško
- MECL, "Master Equipment Component List"

10 CFR 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants [12]

For the preparation of the overall Aging management program NEK used U.S. NRC 10 CFR 54. Parts of 10 CFR 54 [12] applicable for NEK are:

- 54.4 Scope;
- 54.21 Contents of application--technical information.

Regulatory Guide 1.188, Standard Format and Content for License Renewal Application [15]

NEK used RG 1.188, Rev. 1 for the preparation of its license renewal application. This Regulatory Guide has endorsed NEI Guideline 95-10, Revision 6, entitled "*Industry Guideline for implementing the requirements of 10 CFR Part 54 – The License Renewal Rule*".

NUREG-1800, Standard Review Plan - License Renewal (SRP-LR) [13]

The principal purposes of the SRP-LR are to ensure the quality and uniformity of the reviews and to present a well-defined basis from which to evaluate applicant programs and activities for the period of extended operation. It is divided into four major chapters: (1) Administrative information; (2) Scoping and screening methodology for identifying S/Cs subject to aging management review, and implementation results; (3) Aging management review results; and (4) Time-limited aging analyses.

Each SRP-LR section is organized into six subsections. Subsection 1 describes the scope of the review; Subsection 2 contains a statement of the purpose of the review and identification of applicable NRC requirements and the technical basis for determining the acceptability of programs and activities within the area of review; Subsection 3 discusses the way the review is accomplished; Subsection 4 presents the type of conclusion that is sought for the particular review area; Subsection 5 discusses the NRC staff's plan for using the SRP-LR sections; and lastly, Subsection 6 lists the references used in the review process.

The GALL report (NUREG-1801) is referenced as a technical basis document in SRP-LR.

NUREG-1801, Generic Aging Lessons Learned [18]

It identifies aging management programs (AMPs) that were determined to be acceptable to manage the aging effects of S/Cs in the scope of license renewal as required by 10 CFR Part 54.

The GALL report is a compilation of acceptable aging management programs and corrective measures that would be needed for the existing plant programs to become acceptable for S/Cs with specified materials, environments and aging effects. The GALL is the result of a thorough operating experiences review of U.S. and worldwide NPPs and it represents an overall guidance for AMP application in NEK.

NEI 95-10, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule" [16]

The U.S. nuclear industry developed an NEI Guideline 95-10 to provide further guidance to the licensees for implementing the Amended Rule 10 CFR Part 54 and for use of the GALL report in the preparation of license renewal applications. The NRC published Regulatory Guide (RG) 1.188 to endorse the NEI 95-10, Revision. 6. The major elements of this guideline which were all used for the preparation of the NEK AMP include:

a. Identifying the SSCs within the scope of license renewal;
b. Identifying the intended functions of SSCs within the scope of license renewal;
c. Identifying the S/Cs subject to aging management review and intended functions;
d. Assuring that effects of aging are managed;
e. Application of new programs and inspections for license renewal;
f. Identifying and resolving TLAAs; and
g. Identifying a standard format and content of a license renewal application.

USAR, "Updated Safety Analysis Report, Nuclear Power Plant Krško [10]

Updated Safety Analysis Report is the highest rank document of the Krško NPP. It provides evaluated configurations of the plant in all its states.

USAR mechanical systems flow diagrams were used for selecting mechanical components within the scope of AMP.

With establishing the AMP in NEK, the new USAR section 18 - "*NPP Krško Aging Management Program*" is added to NEK USAR.

Technical Specifications (design and/or purchase)

Technical Specifications for equipment, components and structures are used as a source of data which are required for Aging Management Review.

MECL, "Master Equipment Component List" [33]

MECL (Master Equipment Component List) is the main component database, which contains all important components, equipment and structures. This database enables data filtering according to a series of attributes and help get a set of components or a component with specific attributes.

To support AMP the MECL database is extended with cables and structures because it enables use of preventive maintenance program database. This is important because it ensures better transparency, data records and operating experiences collection.

All MECL components or structures that are part of AMP list are designated in the MECL database with AMP flag Y (Yes).

b) Key elements used in plant programs to assess aging

The NEK's aging management program, developed within the regulatory framework that relied on GALL Report [18], was designed to fully comply with industry experience and best practices internationally. Generic AMPs were developed reflecting the best knowledge from the R&D and industry practices and assessed by U.S. NRC as relevant for use in justifying the arrangements to allow for the necessary conditions for the license renewals as defined by the 10 CFR 54. The appropriate AMPs compiled during the establishment of the AMP NEK program are developed to assure that the relevant SSCs are monitored through the extended lifetime, as it is defined in the GALL report [18].

Consequently, the review of plant-specific programs was to verify that those are in compliance with the requirements of the GALL [14] and [18]. In addition, the overall conditions within the AMPs are to be implemented, have to be bounded by the conditions for which the GALL program is applicable.

The review of this element of the NEK AM Project included a systematic evaluation of each of the plant-specific aging management programs identified in the AMR task for its adequacy in managing all aging effects applicable to particular structures and components.

The evaluation of NEK's existing programs addressed all relevant features ('*generic*' attributes) and determined whether they are consistent with the requirements provided in GALL [18].

List of GALL 10 elements/attributes of AMP

Scope of Program
The scope of the program should be defined in terms of S/Cs that are addressed by the program with regard to aging.

Preventive Actions
For programs of prevention and mitigation types, the activities/actions that are proposed should be described. For condition or performance monitoring programs, this information need not be provided.

Parameters Monitored or Inspected
The parameters to be monitored or inspected should be identified and linked to the degradation of the particular S/C intended function(s).
For prevention and mitigation programs, the parameters monitored should be the specific parameters being controlled to achieve prevention or mitigation of aging effects.
For a condition monitoring program, the parameter monitored or inspected should detect the presence and extent of aging effects.
For a performance monitoring program, a link should be established between the degradation of the particular S/C intended function(s) and the parameter(s) being monitored.

Detection of Aging Effects
The parameters to be monitored or inspected should be clearly specified. They should be appropriate to detecting aging effects before there is a loss of the S/C intended function(s). They should ensure that the structure and component intended function(s) will be adequately maintained for license renewal under all CLB design conditions.
Information should be provided on the inspection method or technique (e.g. visual, volumetric, surface inspection), frequency, sample size, data collection and timing of inspections to ensure timely detection of aging effects. Information that links the parameters to be monitored or inspected to the aging effects being managed should also be given.
The program should describe activities to collect data as part of the program (i.e. '*when*', '*where*' and '*how*' program data are collected).
The method or technique and frequency may be linked to plant-specific or industry-wide operating experience. Justification should be provided that the technique and frequency are adequate to detect the aging effects before a loss of S/C intended function.
When sampling is used to inspect a group of SCs, the basis for the inspection population and sample size should be specified. The samples should be oriented on the locations most susceptible to the specific aging effect of concern in the period of extended operation.
Provisions should also be included on expanding the sample size when degradation is detected in the initial sample.

Monitoring and Trending
Monitoring and trending activities should be described. These activities should provide predictability of the extent and rate of degradation and thus facilitate timely corrective or mitigative actions. The parameter or indicator trended should be specified. The methodology for analyzing the inspection or test results against the acceptance criteria should be described. The appropriateness of the technique and frequency should be evaluated based on plant-specific and/or industry-wide operating experience.
This program element describes '*how*' the data collected are evaluated. This may include trending for a forward look. This includes an evaluation of the results against the acceptance criteria and a prediction regarding the rate of degradation in order to confirm that timing of the next scheduled inspection will occur before a loss of S/C intended function.

Acceptance criteria
The acceptance criteria of the program and its basis should be described. The acceptance criteria, against which the need for corrective actions will be evaluated, should ensure that the structure and component intended function(s) are maintained under all CLB design conditions during the period of extended operation. The program should include a methodology for analyzing the results against applicable acceptance criteria.
Acceptance criteria could be specific numerical values, or could consist of a discussion of the process for calculating specific numerical values of conditional acceptance criteria to ensure that the structure and component intended function(s) will be maintained under all CLB design conditions. Information from available references may be cited.

Corrective Actions
Actions to be taken when the acceptance criteria are not met should be described. Corrective actions, including root cause determination and prevention of recurrence, should be timely.

If corrective actions permit analysis without repair or replacement, the analysis should ensure that the structure and component intended function(s) will be maintained consistent with the CLB.

Confirmation Process

The confirmation process should be described. It should ensure that preventive actions are adequate and that appropriate corrective actions have been completed and are effective.

The effectiveness of prevention and mitigation programs should be verified periodically. When corrective actions are necessary, there should be follow-up activities to confirm that the corrective actions were completed, the root cause determination was performed, and recurrence is prevented.

Administrative Controls

Programs relied on to manage aging for license renewal must be administratively controlled. The administrative controls of the program should be described. They should provide a formal review and approval process. Summary description of the program and activities for managing the effects of aging for license renewal should be included in the FSAR/USAR.

Operating experience

Operating experience with existing programs should be discussed. The operating experience of aging management programs, including past corrective actions resulting in program enhancements or additional programs, should be considered. This information can show where an existing program has succeeded and where it has failed (if at all) in intercepting aging degradation in a timely manner. This information should provide objective evidence to support the conclusion that the effects of aging will be managed adequately so that the structure and component intended function(s) will be maintained during the period of extended operation.

An applicant may have to commit to providing operating experience in the future for new programs to confirm their effectiveness.

The 10-elements evaluations of NEK AMPs are documented in the ESD-TR-05/14 [19] report. The GALL 10-elements (attributes) evaluation is essential to determine the adequacy of the program to be used as an aging management program (AMP). All NEK AMP 10-elements evaluations were performed by engineers that are responsible for regular update of programs and procedure. The performance of the assessment and the record keeping is in full compliance with the MD-5 "*Aging Management Program*" requirements.

c) Processes/procedures for the identification of aging mechanisms and their possible consequences;

Detailed evaluation was performed for each mechanical component or civil or electrical commodity group that were determined to be relevant for aging management at NEK. All available records including original Technical Specification, Purchase Orders, Master Equipment List, Design Modification Packages and other documentation were taken into account.

Evaluation considers:

- Plant system a SSC belongs to,
- type of component or commodity,
- material, and
- environment (internal and external) of a component or commodity that is subject to.

A component or a commodity, potential aging mechanisms and relevant AMPs to remedy those were identified in compliance with the GALL Report, NUREG-1801 [18]. Any specific AMP was evaluated according to 10-elements (attributes) listed under b) above. The adequacy of each existing (or indeed modified or newly developed) AMP, to manage aging effects for particular structures and components reflected the 10 elements requirements of the GALL.

To make the confirmation as required by the NUREG-1801 [18] 10 program elements, NEK reflected its choice of a single program/activity or a combination of several programs/activities. Once the approach was defined the potential aging management program/activity was evaluated against the 10 elements of the GALL.

The existing NEK's programs fulfil the requirements of the 10 elements of GALL, Revision 2. A list of the programs is provided in Table 2-4.

Table 2-4: Implementing programs and procedures for AMP activities

NUREG-1801 Number	NUREG-1801, Rev. 2 PROGRAM	NEK Program/Procedure
XI.M1	ASME Section XI In-service Inspection Subsections IWB, IWC and IWD	TD-2E/4; In-service Inspection Program - The 4th Inspection Interval
XI.M2	Water Chemistry	ADP-1.6.021; Kemijske specifikacije in kriteriji za korektivno ukrepanje (Chemical specifications and criteria for corrective actions) CAP-6.001; Plan kemijskega in radiokemijskega vzorčevanja (Chemical and radio-chemical sampling plan)
XI.M3	Reactor Head Closure Studs	TD-2E/4; Inservice Inspection Program - The 4th Inspection Interval
XI.M4	BWR Vessel ID Attachment Welds	Not Applicable to PWR
XI.M5	BWR Feedwater Nozzle	Not Applicable to PWR
XI.M6	BWR Control Rod Drive Return Line Nozzle	Not Applicable to PWR
XI.M7	BWR Stress Corrosion Cracking	Not Applicable to PWR
XI.M8	BWR Penetrations	Not Applicable to PWR
XI.M9	BWR Vessel Internals	Not Applicable to PWR
XI.M10	Boric Acid Corrosion	TD-2J; Boric Acid Inspection Program
XI.M11B	Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components (PWRs Only)	TD-2S; Program nadzora inconela 600/82/182 (Inconel 600/82/182 Surveillance program) TD-2R; Program nadzora glave reaktorske posode in BMI penetracij (Reactor Head and BMI surveillance program) TD-2E/4; Inservice Inspection Program - The 4th Inspection Interval
XI.M12	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	Not Applicable to NEK
XI.M13	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)	Not Applicable to NEK
XI.M14	Loos Part Monitoring	Not Credited
XI.M15	Neutron Noise Monitoring	Not Credited
XI.M16	PWR Vessel Internals	TD-2O; Program nadzora notranjih delov reaktorske posode in mehanizmov kontrolnih palic (Reactor Internals and Control Rod Drive Mechanism Surveillance Program)

NUREG-1801 Number	NUREG-1801, Rev. 2 PROGRAM	NEK Program/Procedure
		TD-2E/4; Inservice Inspection Program - The 4th Inspection Interval
XI.M17	Flow Accelerated Corrosion	QD4; Program Erozije/Korozije (Erossion/Corossion Program)
XI.M18	Bolting Integrity	TD-2E/4; Inservice Inspection Program - The 4th Inspection Interval
XI.M19	Steam Generator Tube Integrity	TD-0H; Program Uparjalnikov (Steam Generator Program)
XI.M20	Open-Cycle Cooling Water System	TD-1Z1; Open-Cycle Cooling Water System
XI.M21	Closed-Cycle Cooling Water System	TD-1Z2; Closed-Cycle Cooling Water System
XI.M22	Boraflex Monitoring	Not Applicable to NEK
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	ADP-1.1.141; Ravnanje s težkimi bremeni v NEK (Heavy Load Handling in NEK) ADP-1.1.142; Uporaba dvigal, dvižnih naprav, viličarjev in pomožnih nosilnih sredstev v NEK (Use of Cranes, lifting devices, fork lifts and auxiliary transport devices in NEK) ADP-1.4.160; Program preventivnega vzdrževanja dvigal, opreme za prenos goriva in pomožnih nosilnih sredstev (Preventive Maintenance program for lifting devices, equipment for fuel transfer and auxiliary transport devices)
XI.M24	Compressed Air Monitoring	ADP-1.4.225; Program preventivnega vzdrževanja kompresorjev (Preventive Maintenance Program for Compressors) GMM-4.550; Splošni postopek strojnega vzdrževanja kompresorjev (Compressor Mechanical Maintenance General Procedure) TD-2E/4; Inservice Inspection Program - The 4th Inspection Interval
XI.M25	BWR Reactor Water Cleanup System	Not Applicable to a PWR
XI.M26	Fire Protection	TD-6; Program požarne zaščite-požarni red (Fire Protection Program)
XI.M27	Fire Water System	QD-5; Program inspekcije protipožarnega sistema (Fire Protection Inspection Program) TD-6; Program požarne zaščite-požarni red (Fire Protection Program)
XI.M28	Buried Piping and Tank Surveillance	Deleted.
XI.M29	Aboveground Steel Tanks	TD-2ZZ; Izvajanje pregledov nadzemnih tankov (Above Ground Tanks Surveillance Program)
XI.M30	Fuel Oil Chemistry	OSP-3.4.308; 3-mesečni test kvalitete goriva za diesel generatorja (Diesel Generator 3-month Fuel Oil Quality Test)
XI.M31	Reactor Vessel Surveillance	ED-5; Reactor Vessel Irradiation Surveillance Program
XI.M32	One-Time Inspection	TD-2XX; One-Time Inspection
XI.M33	Selective Leaching of Materials	TD-2XX; One-Time Inspection

NUREG-1801 Number	NUREG-1801, Rev. 2 PROGRAM	NEK Program/Procedure
XI.M34	Buried Piping and Tanks Inspection	Not Credited
XI.M35	One-Time Inspection of ASME Code Class 1 Small-Bore Piping	TD-2X; One-Time Inspection of ASME Code Class 1 Small-Bore Piping
XI.M36	External Surfaces Monitoring Program	TD-2AA; External Surfaces Monitoring Program
XI.M37	Flux Thimble Tube Inspection	TD-2R; Program nadzora glave reaktorske posode in BMI penetracij (see XI.M11A) (Reactor Head and BMI surveillance program)
XI.M38	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	TD-2W; Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components
XI.M39	Lubrication Oil Analysis	ADP-1.4.130; Program podmazovanja strojne opreme Nuklearne elektrarne Krško (NEK Mechanical Equipment Lubrication Program)
XI.M40	Monitoring Of Neutron-Absorbing Materials Other Than Boraflex	ED-4; Surveillance Program for Borated Stainless Steel Sheets
XI.M41	Buried and Underground piping and Tanks	TD-2Z; Buried and Underground Piping and Tanks
XI.S1	ASME Section XI, Subsection IWE	TD-2H/2; Containment Inspection Program
XI.S2	ASME Section XI, Subsection IWL	Not Applicable to NEK
XI.S3	ASME Section XI, Subsection IWF	TD-2E/4; Inservice Inspection Program - The 4rd Inspection Interval
XI.S4	10 CRF 50, Appendix J	ADP-1.1.235; Containment Leakage Rate Testing Program
XI.S5	Masonry Wall Program	Not Applicable to NEK
XI.S6	Structures Monitoring Program	TD-2L; Program nadzora gradbenih objektov in konstrukcij v NE Krško (Civil Structures Surveillance Program) TD-2N; Program tehničnih opazovanj gradbenih objektov in konstrukcij (Civil Structures Technical Observation Program) TD-2U; Program nadzora nosilnih jeklenih konstrukcij v NE Krško (NEK Steel Structures Surveillance Program)
XI.S7	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	Not Credited
XI.S8	Protective Coating Monitoring Maintenance Program	Not Credited
XI.E1	Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	TD-2D; Cable Aging Management Program ADP-1.4.451; Program preventivnega vzdrževanja nizkonapetostnih elektromotorjev (Low-voltage Motors Preventive Maintenance Program) ADP-1.4.453; Program preventivnega vzdrževanja visokonapetostnih elektromotorjev (High-voltage Motors Preventive Maintenance Program)

NUREG-1801 Number	NUREG-1801, Rev. 2 PROGRAM	NEK Program/Procedure
		ADP-1.4.454; Program preventivnega vzdrževanja MOV aktuatorjev (MOV Actuators Preventive Maintenance Program) ADP-1.4.455; Program preventivnega vzdrževanja nizkonapetostnih stabilnih naprav (Low-voltage Stabile Equipment Preventive Maintenance Program) ADP-1.4.456; Program preventivnega vzdrževanja visokonapetostnih stabilnih naprav (High-voltage Stabile Equipment Preventive Maintenance Program) ADP-1.4.457; Program preventivnega vzdrževanja regulatorjev napetosti in električne zaščite (Voltage Regulator and Over-voltage Protection Preventive Maintenance Program) ADP-1.4.459; Program preventivnega nadzora staranja električnih kablov (Cable Preventive Aging Management Program)
XI.E2	Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	TD-2D; Cable Aging Management Program ADP-1.4.459; Program preventivnega nadzora staranja električnih kablov (Cable Preventive Aging Management Program) ADP-1.4.600; Program nadzornega preverjanja in umerjanja instrumentacije vgrajene v NE Krško (NEK Instrumentation Surveillance and Calibration Program) ADP-1.4.603; Preverjanje, umerjanje in verifikacija instrumentacijske opreme vgrajene v NE Krško (NEK Instrumentation Equipment Surveillance, Calibration and Verification)
XI.E3	Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	TD-2D; Cable Aging Management Program ADP-1.4.453; Program preventivnega vzdrževanja visokonapetostnih elektromotorjev (High-voltage Motors Preventive Maintenance Program) ADP-1.4.456; Program preventivnega vzdrževanja visokonapetostnih stabilnih naprav (High-voltage Stabile Equipment Preventive Maintenance Program) ADP-1.4.457; Program preventivnega vzdrzevanja regulatorjev napetosti in električne zaščite (Voltage Regulator and Over-voltage Protection Preventive Maintenance Program) ADP-1.4.458; Program preventivnega vzdrzevanja elektricnih instalacij in razsvetljave (Electrical Installations and Lighting Preventive Maintenance Program) ADP-1.4.459; Program preventivnega nadzora staranja električnih kablov (Cable Preventive Aging Management Program)
XI.E4	Metal-enclosed Bus	ADP-1.4.456; Program preventivnega vzdrževanja visokonapetostnih stabilnih naprav (High-voltage Stabile Equipment Preventive Maintenance Program)
XI.E5	Fuse Holders	Not Applicable to NEK
XI.E6	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	TD-2D; Cable Aging Management Program ADP-1.4.451; Program preventivnega vzdrževanja nizkonapetostnih elektromotorjev (Low-voltage Motors Preventive Maintenance Program)

NUREG-1801 Number	NUREG-1801, Rev. 2 PROGRAM	NEK Program/Procedure
		ADP-1.4.453; Program preventivnega vzdrževanja visokonapetostnih elektromotorjev (High-voltage Motors Preventive Maintenance Program) ADP-1.4.454; Program preventivnega vzdrževanja MOV aktuatorjev (MOV Actuators Preventive Maintenance Program) ADP-1.4.455; Program preventivnega vzdrževanja nizkonapetostnih stabilnih naprav (Low-voltage Stabile Equipment Preventive Maintenance Program) ADP-1.4.456; Program preventivnega vzdrževanja visokonapetostnih stabilnih naprav (High-voltage Stabile Equipment Preventive Maintenance Program) ADP-1.4.457; Program preventivnega vzdrževanja regulatorjev napetosti in električne zaščite (Voltage Regulator and Over-voltage Protection Preventive Maintenance Program) ADP-1.4.459; Program preventivnega nadzora staranja električnih kablov (Cable Preventive Aging Management Program) ADP-1.4.600; Program nadzornega preverjanja in umerjanja instrumentacije vgrajene v NE Krško (NEK Instrumentation Surveillance and Calibration Program) ADP-1.4.603; Preverjanje, umerjanje in verifikacija instrumentacijske opreme vgrajene v NE Krško (NEK Instrumentation Equipment Surveillance, Calibration and Verification)

d) Establishment of acceptance criteria for aging

The acceptance criteria for the SSCs, subject to aging management are defined by the original design basis for the plant, i.e. during the LTO conditions relevant SSCs (that are included in the aging management program, as defined by 10 CFR 54) are supposed to be performed as in the original design. Consequently any particular Code or Standard that defines performance required for the SSC is identified and/or reflected in the GALL program [18] and its requirements. The details of those acceptance criteria are discussed in the evaluation of individual programs (see chapters 3, 4, 5 and 7).

The acceptance criteria in general could be defined in terms of specific numerical values, or could be a process of determining "conditional acceptance" criteria, ensuring that the SSC's intended function(s) is maintained under all CLB design conditions.

In general, the acceptance criteria shall reflect relevant codes and standards, which are then defining the technical acceptance criteria for the evaluation of the program. Therefore, many provisions of the e.g. ASME Code Section III and XI are referenced in GALL. The IEEE 323-1971 is typically used as the technical basis for evaluation of environmental qualification of electrical components in severe environmental conditions, as required by 10 CFR 50.49 [40]. American Concrete Institute (ACI) Codes and Standards are used as the technical bases for evaluation of concrete structures. Several EPRI reports are used as technical bases for acceptance criteria. These reports are identified related with each specific GALL program. Certain Regulatory Guides and NUREGs addressing particular issues, such as RG. 1.99 [41] and NUREG-0588 [42], are used as technical bases for acceptance criteria. Similarly, these are identified in specific GALL programs.

The acceptance criteria, against which the needs for corrective actions are evaluated, ensure that the structure's or component's intended function(s) are maintained under all CLB design conditions, including, in particular, the period of extended operation (LTO). NEK has in place a Corrective Actions Program, other quality assurance (QA) procedures and administrative controls, all compliant with the requirements of the 10 CFR Part 50, Appendix B, which is required per 10 CFR 54.

e) Use of R&D programs

NEK with its AMP program being developed under and in accordance with the Slovenian regulation, as well as with the US NRC 10 CFR 54, needs to be compliant with all current »*state-of-the-art*" technology. This implies that NEK's AMP program needs to be updated whenever a new e.g. GALL document is issued. The NEK Aging Management Program was initially based on the revision 1 of GALL and later upgraded to the revision 2 of GALL, issued in 2010.

The GALL Report is supported by the following institutes: Electric Power Research Institute (EPRI), Nuclear Energy Institute (NEI), American Concrete Institut (ACI), Institute of nuclear power operations (INPO), American Petroleum Institute (API), and American Institute of Steel Construction (AISC).

NEK is involved in a few R&D programs for cables aging management, as it is described in section 3.1.2.1.

f) Use of internal and external operating experience

Plant-specific and industry operating experience including past corrective actions resulting in process enhancements, was considered in the development of the aging management programs. This information provides objective evidence that the effects of aging have been, and will continue to be, adequately managed. The implementing procedures for the review of operating experience (ADP-1.1.200 "Operating Experience Assessment Program" Ref. [43]) provides an information source for incorporating additional plant-specific and industry operating experience into the aging management programs to ensure continued NEK program effectiveness.

Any observed deficiency at NEK is included into the NEK Corrective Action Program (CAP) database. A group of NEK experts has a task to assign a new corrective action for equipment and include it into CAP system database with attributes, and one of them is also AMP flag (Y – yes or N – no). Any Corrective actions including those with AMP=Y are resolved in accordance with 10 CFR 50, Appendix B. In case the resolution would reveal a new aging mechanism that is not proposed by the AMP then the AMP should be revised to incorporate new aging mechanism.

In frame of NEK personnel organization there is a group of engineers for independent safety evaluations. One of group's tasks is external operating experiences and among them the external operating experiences related to AMP. Based on a wide research of the worldwide NPPs operating experiences, all such potentially NEK applicable operating experiences are included into NEK Corrective Action Program database. The external operating experience is evaluated by the NEK responsible engineer. So far, no NEK or external operating experience evaluation required a change of AMPs.

NEK uses GALL Report as a database of evaluated worldwide operating experiences. NEK confirms that the conditions and operating experience at the plant are bounded by the conditions and operating experience evaluated in the GALL Report.

2.3.3 Monitoring, testing, sampling and inspection activities

The overall Aging Management Program (NEK program MD-5 [1]) defines processes for monitoring, inspection and surveillance activities to assess aging effects to identify unexpected behavior or degradation during service. This section provides summary descriptions of the programs credited for managing the effects of aging.

a) <u>Programs for monitoring condition indicators and parameters and trending</u>

XI.M2 / Water Chemistry / ADP-1.6.021, CAP-6.001
The Water Chemistry program is consistent with revision 2 of NUREG-1801, Section XI.M2, "*Water Chemistry*". It manages the aging effects of cracking, denting, fouling, and loss of material for components that are within the scope of the program. The intent of the Water Chemistry program is to minimize corrosion in order to maintain the system pressure boundary integrity.
The Water Chemistry program relies on the periodic monitoring and control of known detrimental contaminants such as chloride, fluoride, dissolved oxygen and sulphate concentrations below the levels known to result in cracking, loss of material and reduction of heat transfer. Water chemistry control is based on the industry guidelines for primary and secondary water chemistry.
The Water Chemistry program is consistent with the guidelines of EPRI "*PWR Primary Water Chemistry Guidelines*", EPRI "*PWR Secondary Water Chemistry Guidelines*" and Siemens KWU *Secondary Water Chemistry Guidelines*.

XI.M17 / Flow-Accelerated Corrosion / QD4
The Flow-Accelerated Corrosion is consistent with revision 2 of NUREG-1801, Section XI.M17, "*Flow-Accelerated Corrosion*". The Flow-Accelerated Corrosion program manages the aging effect of wall thinning thus assuring that the structural integrity of all steel (carbon or low-alloy) piping and components containing high-energy fluids (two phase as well as single phase) is maintained.
The program applies to both safety-related and non-safety-related components. The Flow–Accelerated Corrosion program is based on "*Recommendations for an Effective Flow-Accelerated Corrosion Program* (NSAC-202L-R3) [44]" and predicts, detects, and monitors FAC in plant piping and other pressure retaining components. The program (a) conducts an analysis to determine critical locations (CHECWORKS), (b) performs limited baseline inspections to determine the extent of wall thinning at those locations, and (c) performs follow-up inspections to confirm the predictions or repairing or replacement of piping and components as necessary. CHECWORKS is a predictive computer program that uses past inspection data to predict wear rates.

XI.M21 / Closed Treated Water Systems / TD-1Z1

The Closed Treated Water Systems program is consistent with revision 2 of NUREG-1801, Section XI.M21, "*Closed Treated Water Systems*". The Closed Treated Water Systems program manages the aging effects of reduction of heat transfer due to fouling, or loss of material from and cracking due to corrosion and/or stress corrosion cracking of the components in the Component Cooling System, Chilled Water Cooling System, and Diesel Generator Cooling System. The Closed Treated Water Systems program manages aging of in-scope components with corrosion control strategies and chemistry specifications, including the use of inhibitors.

XI.M24 / Compressed Air Monitoring / ADP-1.4.225, GMM-4.550, TD-2E/4
The Compressed Air Monitoring program is consistent with revision 2 of NUREG-1801, Section XI.M24, "*Compressed Air Monitoring*". The Compressed Air Monitoring program manages the aging effect of loss of material due to corrosion for the components in the Compressed Air System and Instrument Air System. The Compressed Air Monitoring program performs air quality sampling visual inspections, and periodic testing to verify the adequacy of the air quality and to detect air leakage.

XI.M27 / Fire Water System / QD-5, TD-6
The Fire Water System Program is consistent with revision 2 of NUREG-1801, Section XI.M27, "*Fire Water System*". The Fire Water System program manages the aging effect of loss of material and fouling that leads to corrosion in fire protection piping systems. The Fire Water System Program performs periodic flushing, system performance testing and inspections. Many of these systems are normally maintained at required operating pressure and monitored in such a way that leakage resulting in loss of system pressure is immediately detected and corrective actions initiated.

XI.M29 / Aboveground Metallic Tanks / TD-2ZZ
The Above Ground Steel Tanks Program is consistent with revision 2 of NUREG-1801, Section XI.M29, "*Above Ground Steel Tanks*". The Above Ground Steel Tanks program manages loss of material of external surfaces of above ground carbon steel tanks by periodic visual inspection of external surfaces and thickness measurement of locations that are inaccessible for external visual inspection.

XI.M30 / Fuel Oil Chemistry / OSP-3.4.308
The Fuel Oil Chemistry program is consistent with revision 2 of NUREG-1801, Section XI.M30, "*Fuel Oil Chemistry*". The Fuel Oil Chemistry program manages the aging effect of loss of material and fouling that leads to corrosion on piping and components in the underground diesel fuel tanks and transfer system by maintaining potentially harmful contaminants at low concentrations. The program manages loss of material at locations where contaminants can accumulate, such as through thickness measurement of the tank bottom.

XI.M31 / Reactor Vessel Surveillance / ED-5
The Reactor Vessel Surveillance program is consistent with revision 2 of NUREG-1801, Section XI.M31, "*Reactor Vessel Surveillance*". The Reactor Vessel Surveillance program includes the applicable limitations on operating conditions to which the surveillance capsules were exposed (e.g. neutron flux, spectrum, irradiation temperature, etc.). The program also includes ex-vessel monitoring and requirements for storing and possible recovery of tested and untested capsules.
The Reactor Vessel Surveillance program manages the aging effects of loss of fracture toughness due to irradiation embrittlement of the reactor pressure vessel low alloy steel material. Monitoring methods are in accordance with 10 CFR 50, Appendix H. This program includes surveillance capsule removal and specimen mechanical testing/evaluation, radiation analysis and development of pressure-temperature limits. The program ensures, that the reactor vessel materials meet the fracture toughness requirements of 10 CFR 50, Appendix G, and the requirements of Pressurized Thermal Shock (PTS) and upper shelf energy in 10 CFR 50.60 and 10 CFR 50.61.

XI.M33 / Selective Leaching/ TD-2XX
The Selective Leaching program is consistent with revision 2 of NUREG-1801, Section XI.M33, "*Selective Leaching*". The Selective Leaching program manages the aging effects of loss of material on internal and external surfaces of in-scope components made of cast iron and copper alloys (>15% Zn or >8% A). The program performs a one-time inspection of selected components within the scope of the program for loss of material due to selective leaching. The inspection utilizes techniques that are commensurate with industry standard techniques available for selective leaching detection at the time the inspection is performed. The Selective Leaching Program inspection should be completed within 5-year period prior to the period of extended operation which starts in year 2023.

XI.M36 / External Surfaces Monitoring of Mechanical Components / TD-2AA
The External Surfaces Monitoring of Mechanical Components program is consistent with revision 2 of NUREG-1801, Section XI.M36, "*External Surfaces Monitoring of Mechanical Components*". The External Surfaces Monitoring of Mechanical Components program manages the aging effect of loss of material, cracking and change in material properties by visually inspecting the external surfaces of in-scope components, piping, supports, structural members, and structural commodities whether they are constructed of metal or elastomers. External surface visual inspection is performed within this program.

XI.M37 / Flux Thimble Tube Inspection / TD-2R
The Flux Thimble Tube Inspection program is consistent with revision 2 of NUREG-1801, Section XI.M37, "*Flux Thimble Tube Inspection*". The Flux Thimble Tube Inspection program manages the aging effect of loss of material of the flux thimble tube wall. The Flux Thimble Tube Inspection is a condition monitoring program used to inspect for thinning of the flux thimble tube wall, which provides a path for the incore neutron flux monitoring system detectors and forms part of the reactor coolant system (RCS) pressure boundary.
Inspections, techniques, scope and frequency of BMI penetrations are based on the requirements of NRC Bulletin 88-09 "*Thimble Tube Thinning in Westinghouse Reactors*" and NRC Bulletin 2003-02 "*Leakage from Reactor Pressure Lower Head Penetration and Reactor coolant pressure boundary integrity*".

XI.M39 / Lubrication Oil Analysis / ADP-1.4.130
The Lubricating Oil Analysis program is consistent with revision 2 of NUREG-1801, Section XI.M39, "*Lubricating Oil Analysis*". The Lubricating Oil Analysis program manages the aging effects of loss of material and reduction of heat transfer due to fouling for components within the scope of the program. The Lubricating Oil Analysis program maintains oil system contaminants (water and particulates) within acceptable limits, thereby preserving an environment that is not conducive to loss of material or reduction of heat transfer due to fouling.

XI.M41 / Buried and Underground Piping and Tanks / TD-2Z
The Buried and Underground Piping and Tanks program is consistent with revision 2 of NUREG-1801, Section XI.M41, "*Buried and Underground Piping and Tanks* ". The "Buried and Underground Piping and Tanks" program manages the aging effect of loss of material for the underground components within the scope of the program. The program performs periodic visual inspections of the external surface of a representative sample of the material/protective measures combinations of the in-scope buried piping and components.
The program inspects for evidence of damaged wrapping; coating defects such as coating perforation, voids or other damage; and evidence of loss of material on the external surface of the piping or component.

XI.S1 / ASME Section XI, Subsection IWE / TD-2H/2
The ASME Section XI, Subsection IWE program is consistent with revision 2 of NUREG-1801, Section XI.S1, "*ASME Section XI, Subsection IWE*". NUREG 1801, Section XI.S1 specifies the use of ASME Section XI, Subsection IWE, 2004 Edition. NEK is using ASME Section XI 2007 Edition with 2008 Addenda. Use of the 2007 Code with 2008 Addenda is consistent with provisions in 10 CFR 50.55a to use the ASME Code in effect 12 months prior to the start of the inspection interval.
The ASME Section XI, Subsection IWE program manages aging effects of loss of material, cracking, loss of leak tightness, and loss of preload in the reactor containment. The ASME Section XI, Subsection IWE program consists of examinations of metal pressure boundary surfaces and welds, penetrations, integral attachments and their welds, moisture barriers, and pressure-retaining bolted connections.

XI.S4 / 10 CFR 50, Appendix J / ADP-1.1.235
The 10 CFR 50, Appendix J program is consistent with revision 2 of NUREG-1801, Section XI.S4, "*10 CFR Part 50, Appendix J*". The Reactor Containment Leakage Testing 10 CFR 50, Appendix J program manages the aging effects of cracking, loss of leak tightness, loss of preload, and loss of material. The program performs leak test through the Reactor Containment including the systems penetrating the Reactor Containment, penetrations, isolation valves, fittings and access openings to detect degradation of the containment pressure boundary. The 10 CFR 50, Appendix J program is implemented using Option B, as defined and described in NRC Regulatory Guide 1.163, "*Performance-Based Containment Leak-Test Program*", and NEI 94-01, "*Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J*".

XI.S6 / Structures Monitoring Program / TD-2L, TD-2N, TD-2U
The Structures Monitoring Program is consistent with revision 2 of NUREG-1801, Section XI.S6, "*Structures Monitoring*". The Structures Monitoring Program manages the aging effects in material properties for concrete and steel structures. The program relies on periodic visual inspections to monitor the condition of structures, structural elements (including component supports) and miscellaneous structural commodities.

b) Inspection programs

XI.M1 / ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD Program / TD-2E/4
The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program is consistent with revision 2 of NUREG-1801, Section XI.M1, "*ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD*". The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program manages the aging effects of cracking, loss of fracture toughness and loss of material for the ASME Class 1, 2 and 3 piping and components. Program is based on ASME Section XI, Edition 2007, Addenda 2008.

ISI program employs examination techniques as described in ASME Section XI and ASME Section V such as Visual Examination Method, Surface Examination Method including Magnetic Particle, Liquid Penetrant and Eddy Current and Volumetric Examination Method including Ultrasonic, Radiographic, Eddy Current and Acoustic Emission Examinations.
The inspection intervals are in accordance with Articles IWA-2400, IWB-2400, IWC-2400, IWD-2400, and IWF-2400 of ASME Section XI Code. Each interval has a nominal 10-year duration.
The examination categories and requirements are in accordance with Articles IWB-2000, IWC-2000, IWD-2000 and IWF-2000 of ASME Section XI. Examination Category B-J, B-F, C-F-1 and C-F-2 for piping welds are reassigned in accordance with ASME Code Case N-578-1 and N-770-2. The evaluation of non-destructive examination results is in compliance with Article IWA-3000. Components or parts with indications exceeding acceptance standards are accepted either by analytical evaluation or by Repair/Replacement activity.

XI.M3 / Reactor Head Closure Stud Bolting / TD-2E/4
The Reactor Head Closure Stud Bolting program is consistent with revision 2 of NUREG-1801, Section XI.M3, "*Reactor Head Closure Stud Bolting*". NUREG 1801, Section XI.M3 specifies the use of ASME Section XI, Subsection IWB, 2004 Edition. The NEK fourth interval ISI Program uses ASME Section XI 2007 Edition with 2008 Addenda as modified by 10 CFR 50.55a and approved code cases. Use of the 2007 Code with 2008 Addenda is consistent with provisions in 10 CFR 50.55a to use the ASME Code entering into effect 12 months prior to the start of the inspection interval.
The Reactor Head Closure Stud Bolting program manages the aging effects of cracking and loss of material for the reactor head closure stud assembly including nuts and washers. The program includes preventive measures to mitigate cracking and loss of material and visual or volumetric examinations to monitor this degradation. The preventive measures implemented by the program are consistent with the measures identified in NRC Regulatory Guide 1.65, "*Material and Inspection for Reactor Vessel Closure Studs*". The Reactor Head Closure Stud Bolting program visual and volumetric examinations are performed in accordance with the ASME Section XI 2007 Code Edition through the 2008 Addenda, Examination Category B-G-1.

XI.M10 / Boric Acid Corrosion / TD-2J
The Boric Acid Corrosion program is consistent with revision 2 of NUREG-1801, Section XI.M10, "*Boric Acid Corrosion*". The Boric Acid Corrosion program manages the aging effect of loss of material for the in-scope systems, structures and components that are subject to borated water leakage.
The program performs visual inspections to identify boric acid leakage. The scope of the program includes those systems and components which are potential sources of borated water leakage and potential targets of borated water leakage. Generic Letter 88-05, "*Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants*" and industry guidance are used as reference documents for providing guidance for evaluating the severity of boric acid leakage and for determining the appropriate corrective actions. The Boric Acid Corrosion program is supported by the inspection opportunities of other programs including inspections performed during in-service inspection pressure tests and inspections performed during power operation and immediately following a unit shutdown.

XI.M11B / Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components / TD-2S, TD-2R, TD-2E/4
Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components program is consistent with revision 2 of NUREG-1801, Section XI.M11B, "*Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components*". The program addresses the issue of cracking of nickel-alloy components (Alloy 600, 82/182) and loss of material due to boric acid-induced corrosion in susceptible, safety-related components in the vicinity of nickel-alloy reactor coolant pressure boundary components.
NEK applies augmented inspections in accordance with 10 CFR 50.55a and NUREG-1801 based on following ASME Code Cases:

- N-722-1, Additional Inspections for PWR Pressure Retaining Welds in Class 1 Pressure Boundary Components Fabricated with Alloy 600/82/182 Materials.
- N-729-4, Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1

- N-770-2, Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material with or Without Application of Listed Mitigation Activities, Section XI, Division 1.

The impacts of boric acid leakage from non-nickel alloy reactor coolant pressure boundary components are addressed in XI.M10, "*Boric Acid Corrosion*". The Water Chemistry program for PWRs relies on monitoring and control of reactor water chemistry based on industry guidelines as described in XI.M2, "*Water Chemistry*". The continued implementation of the program »*Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components*« provides reasonable assurance that aging effects will be managed in such a way that the components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

XI.M16A / PWR Vessel Internals / TD-2O, TD-2E/4

The PWR Vessel Internals program is consistent with revision 2 of NUREG-1801 (GALL), Section XI.M16A, "*PWR Vessel Internals*". The program is developed according to the comprehensive generic industry program, entitled Materials Reliability Program (MRP-227), being performed by EPRI. The NEK intends to consistently apply all recommendations for the aging management program of the reactor internals components.

The PWR Vessel Internals program manages the following aging effects:

- loss of material due to wear,
- loss of fracture toughness due to neutron irradiation embrittlement,
- loss of preload due to thermal and irradiation enhanced stress relaxation,
- cracking due to stress corrosion cracking,
- cracking due to irradiation-assisted stress corrosion cracking,
- change in dimension due to void swelling,
- cracking due to fatigue,
- fatigue.

Acceptance criteria are provided in MRP-227, ASME Section XI 2007 Code Edition through the 2008 Addenda and in WCAP-17096-NP.

These examinations provide reasonable assurance that the effects of age-related degradation mechanisms will be managed during the period of extended operation. The program includes expanding periodic examinations and other inspections if the extent of the degradation effects exceeds the expected levels.

The continued implementation of the PWR Vessel Internals program provides reasonable assurance that aging effects will be managed in such a way that the systems and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

XI.M18 / Bolting Integrity / TD-2E/4

The Bolting Integrity program is consistent with revision 2 of NUREG-1801, Section XI.M18, "*Bolting Integrity*". Section XI.M3 (incorporated by reference to Section XI.M18) specifies the use of ASME Section XI, Subsection IWB, 2004 Edition. The NEK fourth interval ISI Program uses ASME Section XI 2007 Edition with 2008 Addenda as modified by 10 CFR 50.55a and approved code cases. Use of the 2007 Code with 2008 Addenda is consistent with provisions in 10 CFR 50.55a to use the ASME Code entering into effect 12 months prior to the start of the inspection interval.

The Bolting Integrity program manages the aging effects of cracking, loss of material and loss of preload for bolting/fasteners. The Bolting Integrity program is consistent with the recommendations for a comprehensive bolting integrity program as delineated in NUREG-1339, "*Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants*", and industry recommendations as delineated in the Electric Power Research Institute (EPRI) NP-5769, "*Degradation and Failure of Bolting in Nuclear Power Plants*", with the exceptions noted in NUREG-1339. The program addresses bolting associated with pressure boundary, mechanical and high strength bolting for component supports.

Maintenance procedures provide detailed instructions for removal and installation of bolted pressure boundary closures, and provide generic guidance on proper bolting practices.

XI.M19 / Stem Generators / TD-0H
The Steam Generators program is consistent with revision 2 of NUREG-1801, Section XI.M19, "*Steam Generators*". The Steam Generators program manages the aging effects of cracking and loss of material for the primary and secondary-side steam generator components fabricated of nickel alloys, stainless steel and steel. The program is based on Technical Specification requirements, meets the intent of NEI 97-06, "*Steam Generator Program Guidelines*" and is credited for aging management of the tubes, tube plugs, tube sleeves, tube supports and secondary-side components whose failure could prevent the steam generator from fulfilling its intended safety function. Acceptance criteria for inspections performed in accordance with the Steam Generators program are based on applicable regulations and standards. Corrective actions for conditions that are adverse to quality are performed in accordance with the Corrective Action Program as a part of the Quality Assurance Program.

XI.M23 / Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems / ADP-1.1.141, ADP-1.1.142, ADP-1.4.160
The Inspection of Overhead Heavy Load and Refueling Handling Systems program is consistent with revision 2 of NUREG-1801, Section XI.M23, "*Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems*". The Inspection of Overhead Heavy Load and Refueling Handling Systems program manages the aging effect of loss of material due to general corrosion and rail wear for the in-scope steel cranes, trolleys, bridges and rails. The program also manages the effects of loss of preload of bolted connections. The program is implemented through periodic visual inspections of the crane, trolley, bridge and rail structural members.

XI.M26 / Fire Protection / TD-6
The Fire Protection program is consistent with revision 2 of NUREG-1801, Section XI.M26, "*Fire Protection*". The Fire Protection program manages the aging effects of change in material properties, cracking, and loss of material for the fire protection components and features. The program performs visual inspections of fire barriers, fire barrier penetrations and seals, fire barrier expansion joints, doors and fire wraps.

XI.M32 / One Time Inspection / TD-2XX
The One-Time Inspection program is consistent with revision 2 of NUREG-1801, Section XI.M32, "*One-Time Inspection*". The One-Time Inspection Program provides additional assurance, through sampling inspections using nondestructive examination (NDE) techniques, that aging is not occurring or that the rate of degradation is so insignificant that additional aging management actions are not warranted. The program includes measures to verify the effectiveness of other aging management programs, such as the Water Chemistry Program, Fuel Oil Chemistry and Lubricating Oil Analysis Program to mitigate aging effects. In other cases this program confirms that a separate aging management program is not warranted when significant aging is not expected to occur. If identified aging effects could adversely impact an intended function, additional actions will be taken to correct the condition, perform additional inspections and/or perform periodic inspections as needed.
The program elements included: (a) determination of the sample size based on an assessment of materials of fabrication, environment, plausible aging effects and operating experience; (b) identification of inspection locations in the system, component or structure based on the aging effect; (c) determination of the examination technique including acceptance criteria that would be effective in managing the aging effect that is being examined; and (d) evaluation of the need for follow-up examination if degradation is identified that could jeopardize an intended function. The program relies on the results of inspections performed during the 10-year inspection interval completed prior to the plant extended life.

XI.M35 / One-Time Inspection of ASME Code Class 1 Small-Bore Piping / TD-2X
The One-Time Inspection of ASME Code Class 1 Small-Bore Piping program is consistent with revision 2 of NUREG-1801, Section XI.M34, "*One-Time Inspection of ASME Code Class 1 Small-Bore Piping*". This program is applicable to small bore ASME Code Class 1 piping with less than 4 inches nominal pipe size (NPS 4), which includes pipe, fittings and branch connections. The ASME Code does not require volumetric examination of Class 1 small bore piping. The One-Time Inspection of ASME Code Class 1 Small Bore

Piping program will manage cracking through the use of volumetric examinations. The program includes a sample selected based on susceptibility, inspectability, dose considerations, operating experience, and limiting locations of the total population of ASME Code Class 1 small bore piping locations. Any evidence of aging effects revealed by the one-time inspection results in an evaluation of the inspection results to determine appropriate corrective actions via the corrective action program. The one-time inspection should be completed within 6-year period prior to the period of extended operation which starts in year 2023.

XI.M38 / Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components / TD-2W
The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components program is consistent with revision 2 of NUREG-1801, Section XI.M38, "*Inspection of Internal Surfaces of Miscellaneous Piping and Ducting Components*". The Inspection of Internal Surfaces program for metallic components manages the aging effects of loss of material, cracking and fouling that leads to corrosion by inspecting the internal surfaces of in-scope metalic components. For in-scope elastomeric components the program manages the aging effects of surface cracking, crazing, scuffing, dimensional change (e.g. "ballooning" and "necking"), shrinkage, discoloration, hardening, loss of strength and loss of material (due to wear). These inspections are opportunistic inspections that are performed during the conduct of maintenance when the system is open and the internal surfaces are available to be inspected.

XI.E1 / Insulation Material for Electrical Cables and Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements / TD-2D, ADP-1.4.451, ADP-1.4.453, ADP-1.4.454, ADP-1.4.455, ADP-1.4.456, ADP-1.4.457, ADP-1.4.459
The Non-EQ Electrical Cables and Connections program is consistent with revision 2 of NUREG-1801, Section XI.E1, "*Insulation Material for Electrical Cable and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements*". The Non-EQ Electrical Cables and Connections program manages the aging effects of reduced insulation resistance and electrical failure of accessible non-EQ electrical cables and connections within the scope of the program. The program visually inspects a sample of accessible electrical cables and connections insulation installed in adverse localized environments. The inspection procedures define the visually inspections for surface anomalies such as embrittlement, discoloration, cracking, swelling, melting or surface contamination due to heat, radiation, moisture or vibration.

XI.E2 / Insulation Material for Electrical Cables and Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits / TD-2D, ADP-1.4.459, ADP-1.4.600, ADP-1.4.603
The Non-EQ Electrical Cables and Connections in Instrumentation Circuits program is consistent with revision 2 of NUREG-1801, Section XI.E2, "*Insulation Material for Electrical Cable and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits*".
The Non-EQ Electrical Cables and Connections in Instrumentation Circuits program manages the aging effects of reduced insulation resistance and electrical failure of accessible non-EQ electrical cables and connections used in instrumentation circuits with sensitive, high-voltage, low-level current signals within the scope of the program that are subject to an adverse localized environment. The program performs a proven cable system test for detecting deterioration of the insulation (such as insulation resistance tests, time domain reflectometry tests or other testing judged to be effective in determining cable insulation condition) for those electrical cables and connections disconnected during calibration, or reviews the results and findings of calibrations for those electrical cables that remain connected during the calibration process. Additionally, program defines evaluation of surveillance results or testing on a sample of instrument cables and connections. The inspection procedures define the visually inspections for surface anomalies such as embrittlement, discoloration, cracking, swelling, melting or surface contamination due to heat, radiation, moisture or vibration.

XI.E3 / Inaccessible Power Cables Not Subject To 10 CFR 50.49 Environmental Qualification Requirements / TD-2D, ADP-1.4.453, ADP-1.4.456, ADP-1.4.457, ADP-1.4.458, ADP-1.4.459
The Inaccessible Power Cables program is consistent with NUREG-1801, Section XI.E3, "*Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements*". This program manages the aging effects of localized damage and breakdown of insulation leading to electrical failure for non-EQ, inaccessible

power cables within the scope of the program that are subject to adverse localized environment, caused by exposure to significant moisture simultaneously with significant voltage. The inspection procedure includes monthly inspection of cable manholes for water collection to prevent inaccessible cables being exposed to significant moisture and draining the water if needed.
For medium voltage and sample of inaccessible low voltage power cables one or more following testing techniques are used: Dielectric Loss (Dissipation Factor/Power Factor), Insulation Resistance, Polarization Index, Line Resonance Analysis, Partial Discharge and AC Voltage Withstand, or other testing that is state-of-the-art at the time the tests will be performed.
Additionally, the program used visual inspection for surface anomalies, such as: embrittlement, discoloration, cracking, swelling, melting, water treeing or surface contamination due to moisture, radiation and heat.

XI.E4 / Metal Enclosed Bus / ADP-1.4.456
The Metal Enclosed Bus program is consistent with revision 2 of NUREG-1801, Section XI.E4, "*Metal Enclosed Bus*". The program performs low resistance measurements, inspections, and thermography examinations. The Metal Enclosed Bus (MEB) program manages the aging effects of reduced insulation resistance, electrical failure and loosening of bolted connections for metal enclosed bus and internal components within the scope of the program. The program uses a sampling methodology whereby sections of the in-scope MEB are visually inspected for cracks, corrosion, foreign debris, excessive dust buildup, evidence of water intrusion and visual inspection of component insulation for surface anomalies, such as discoloration, cracking, chipping or surface contamination.

XI.E6 / Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements / TD-2D, ADP-1.4.451, ADP-1.4.453, ADP-1.4.454, ADP-1.4.455, ADP-1.4.456, ADP-1.4.457, ADP-1.4.459, ADP-1.4.600, ADP-1.4.603
The Electrical Cable Connections program is consistent with revision 2 of NUREG-1801, Section XI.E6, "*Insulation Material for Electrical Cable Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements*". The program manages the aging effect of loosening of electrical connections for non-EQ electrical cable connections within the scope of the program. The program performs a periodic inspection, on a sampling basis, to confirm the absence of loosening of electrical connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion and oxidation. Testing may include thermography, contact resistance testing or other appropriate testing methods.

c) <u>Surveillance programs where appropriate</u>

XI.M40 / Monitoring of Neutron-Absorbing Materials other than Boraflex / ED-4
The Surveillance Program for Borated Stainless Steel (BSS) Sheets is consistent with of NUREG-1801, XI.M40 - "*Monitoring of Neutron-Absorbing Materials other than Boraflex*".
The Surveillance Program for Borated Stainless Steel (BSS) Sheets manages the aging effects of loss of material due to corrosion and reduction of neutron absorbing capacity for BSS Sheets in spent fuel storage racks. For that purpose, BSS sheets (coupons) attached on spent fuel pool racks, are used.
The Surveillance Program for BSS Sheets contains the scope, type, frequency of inspection, and acceptance criteria for neutron absorbing BSS Sheets in storage racks.
The Surveillance program is implemented to assure that degradation of the neutron-absorbing material used in spent fuel pools that could compromise the criticality analysis will be detected. This aging management program (AMP) relies on periodic inspection, testing, monitoring, and analysis of the criticality design to assure that the required 5% sub-criticality margin is maintained during the period of license renewal.

d) <u>Any provisions for identifying any unexpected aging degradation</u>

So far the NEK AMP has not documented any unexpected aging degradation. NEK confirms that all aging degradations that have been found are consistent with the NUREG-1801, Revision 2.

2.3.4 Preventive and remedial actions

All major structures and components are under individual surveillance to timely detect any degradation that can cause losing of the intended function. So far, NEK has performed a few important preventive and remedial actions:

- Replacement of both steam generators in year 2000 due to generic degradation of the Westinghouse SGs pipes;
- Weld Overlay: The weld overlays were installed over the dissimilar metal welds (DMW) of pressurizer nozzles to improve resistance to PWSCC. Process consists of applying an annulus of Alloy 52-type weld material on the outside of a circular geometry component over the susceptible Alloy 82/182 DMW;
- Replacement of components with materials more resistant to PWSCC: Reactor vessel closure head was replaced in NEK preventively, without any flaw indications in Alloy 600/82/182 welds of the closure head. The new head does not contain Alloy 600/82/182 components.

2.4 Review and update of the overall AMP

a) <u>How licensee audit and inspection findings are implemented</u>

In case the acceptance criteria are not met, a corrective action is generated. The Corrective Action Program is implemented in accordance with the requirements of 10 CFR 50, Appendix B [36]. A single corrective actions process is applied regardless of the safety classification of the structure or component. Corrective actions are implemented through the initiation of a corrective action report in accordance with plant procedures for actual or potential problems, including unexpected plant equipment degradation, damage, failure, malfunction or loss. Site documents that implement aging management programs will direct that a corrective action report is prepared in accordance with those procedures whenever non-conforming conditions are found (i.e., the acceptance criteria are not met).

Equipment deficiencies are corrected through the initiation of a work order in accordance with plant procedures. Plant procedures require that a corrective action report also be initiated when equipment deficiencies or the need for corrective maintenance is identified.

All audits, inspections, surveillances, monitoring, and reviews of the AMPs are evaluated for the potential improvement of the effectiveness of the AMP. Based on recommendations, NEK prepares the implementation action plan that is then included in the NEK Corrective Action Program system [38]. In 7 years since the AMP was established, NEK AMP was, within Corrective Action Program, updated 4 times:

- As a result of the independed review of the NEK AMP [2] at the beginning of the year 2011 the AMR project was updated. The 2011 update of AMP structures and components list is based on the MECL database with freeze date of 27th of March, 2011. Independent review discovered 15 inconsistencies, and all of them were implemented during the 2011.
- The Regulatory Authority (SNSA) performed thorough review of the NEK AMP during the 2011. SNSA found 15 deficiencies, and all of them were implemented during the 2011.
- Based on Aging Management Program Evaluation Report, ESD TR-05/14, Rev.1 [19], the NEK aging management program is updated to be consistent with the revision 2 of NUREG-1801. All 10 inconsistencies were implemented before April, 2016.
- Based on 5-year AMP review according to MD-5 program (NEK ESD TR-19/15 [45]) 8 deficiency were included into corrective action plan and implemented during 2017.

b) <u>Evaluation of plant specific and others' operating experiences</u>

The NEK OE database is a Corrective Action Program (ADP-1.0.020 "Corrective Action Program Application", Ref. [38]). An attribute for "*AMP degradation*" was included in CAP process procedure in the beginning of 2013. With the help of the Corrective Action Program reports, the OEs on components and structures with designation "*AMP degradation*" are reviewed and evaluated in frame of a 5-year AMP review required according to MD-5 program. The Corrective Actions Program, quality assurance (QA) procedures, site review and approval process and administrative controls are implemented in accordance with 10 CFR Part 50, Appendix B.

NEK evaluates industry's operating experience issued in NRC Information Notices (IN) and Regulatory Issue Summary (RIS). More and more plants in the US are entering in the extended operation and monitoring the success of the aging management programs is becoming even more important. NEK will enter the extended operation in 2023, and will make the best use of the experience gained by other plants and introduce modifications where necessary in its own AMP programs. Events such as the one of September 2, 2011, where corrosion residues blocked flow in the fire protection system at Monticello Nuclear Generating Plant, Unit 1, illustrate the importance of continuous maintenance of the effectiveness of aging management programs and activities. The NEK evaluation of such events is performed within the Corrective Action Program. NEK regularly informs the Regulatory Authority (SNSA) on the actions performed.

Worldwide industry AMP OE, such as U.S. NRC IN and RIS are important for NEK. They are used as a reliable source of information on potential consequences of aging mechanism and their impact on safe operation of the plant. The responsible engineer for applicable NEK AMP evaluates these information notes and all important industry OE cases are included in the NEK AMP.

NEK has used GALL to identify the applicable aging effects requiring management (AERMs) and the corresponding AM Programs (AMPs). Identification of AERMs and corresponding AMPs is conducted based on the existing industry experience compiled by the U.S. NRC in the form of '*generic*' tables (GALL Report [18]). NEK aging related operating experiences have so far confirmed that all aging mechanisms found were according to the GALL.

c) <u>Evaluation of plant modifications that might influence the overall aging management program</u>

The main activity that needs to be regularly performed is updating the list of equipment from EAM-MECL (Master Equipment and Component List) that is modified with any plant design change. This is performed with the procedure ESP-2.624 – "*Design Impact Evaluation*" [46].

Each plant modification is evaluated for:

- Modified list of EAM-MECL components gives a list of AMP components that might be added or deleted to the existing EAM-MECL list and plant specific programs and procedures related to AMP shall be updated. Each modification gets a flag if this is AMP related or not. The criteria for such designation (AMP=YES) will be any change in EAM-MECL on equipment with flag (AMP=YES);
- Each modification involves potential changes in the procedures, programs and other AMP related activities.

The screening of affected EAM-MECL components is a part of each modification. The responsible engineer for each modification should perform initial screening of affected AMP equipment (changes in EAM-MECL list with AMP=YES) and update will be performed regularly within the 5-year AMP review according to MD-5 – "*Aging Management Program*".

d) <u>Evaluation and measurement of the effectiveness of aging management</u>

Overall NEK's aging management program MD-5 integrated 36 specific AMPs, which are all consistent with NUREG-1801, Revision 2 requirements. All these 36 programs listed in Tables 2-4 have been implemented. The aging mechanisms considered in AMPs are conservatively defined and aging degradations are not very usual to be found on the field. The effectiveness of the aging management is not easy to be

measured, so a better approach is to evaluate it. Aging management program (AMP) effectiveness is already continuously evaluated as part of operating experience (OE) review programs (site-specific and industry-wide) and the corrective action program (CAP).
Every aging management program is evaluated for effectiveness as part of the existing regulatory commitments and as recommended by NEI 95-10, Rev. 6 [16], and NUREG-1800 Standard Review Plan (SRP), Rev. 2 [17]. SRP Branch Technical Position A.1.2.3, "Aging Management Program Elements" describes Element 8, *Confirmation Process*, as central to the ongoing assessment of program effectiveness. This is one of ten program elements integral to every program.
All audits, inspections, surveillances, monitoring, reviews, internal and external operating experiences of the AMPs are evaluated for the potential improvement of the effectiveness of the AMP.

e) <u>Evaluation of aging analyses that are time limited</u>

Time Limited Aging Analyses (TLAA) are plant-specific analyses or calculations based on an explicitly assumed time of plant operation or design life. The criteria that should be used to compile the list of analyses that need to be considered as TLAAs are based on Standard Review Plan for License Renewal, NUREG-1800, Rev. 1 [14], and 10 CFR 54 – "*Requirements for Renewal of Operating Licenses for Nuclear Power Plants*" [12]. The TLAAs are the plant-specific analyses that fulfil all the following conditions:

(1) Involve SSCs within the scope of license renewal as delineated in 10 CFR 54.4(a);
(2) Considers the effects of aging;
(3) Involve time-limited assumptions defined by the current operation term;
(4) Were determined to be relevant by the licensee in making safety determination;
(5) Involve conclusions or provide the basis for conclusions related to the capability of the SSC to perform its intended function(s) as delineated in 10 CFR 54.4(b);
(6) Are contained or incorporated by reference in the current licensing basis (CLB).

According to the 10 CFR Part 54 (c) the evaluation of time-limited aging analyses has been performed and demonstrated that:

i. The analyses remain valid for the period of extended operation;
ii. The analyses have been projected to the end of the period of extended operation; or
iii. The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

TLAAs that were part of Aging Management Review Project are the following:

- AMP-TA-01 / Review of Krško NPP Plant Specific TLAAs Related to Civil Structures
- AMP-TA-02 / Review of NPP Krško Environmental (EQ) Program
- AMP-TA-03 / Screening of Potential TLAA in Krško NPP CLB
- AMP-TA-04 / Krško TLAA RCL Piping and RCS Components Fatigue
- AMP-TA-05 / Krško TLAA Auxiliary Class 1/2/3 Piping Fatigue
- AMP-TA-06 / Krško TLAA Environmental Fatigue Evaluations per NUREG/CR-6260
- AMP-TA-07 / Krško TLAA Reactor Vessel Beltline Fluence Evaluation
- AMP-TA-08 / Krško TLAA Reactor Pressure Vessel Irradiation Embrittlement
- AMP-TA-09 / Krško TLAA Impact of Thermal Aging on Stainless Steel Welds and Cast Material (RCL, Auxiliary Lines, Pump Casing)
- AMP-TA-10 / Krško TLAA Update of USAR Chapters 11 and 15
- AMP-TA-11 / Krško TLAA Class 2/Class 3 Primary Sampling System Lines Fatigue Analysis
- AMP-TA-12 / Krško TLAA Analysis for Containment Penetrations Fatigue Analysis

The review [2] confirmed that the TLAA task was conducted consistently with the adopted methodology. Generally, AMP-TA-01 through AMP-TA-12 documents were found acceptable from the viewpoint of methodology, engineering judgment and industry experience on TLAA.

The first three TLAA related reports, AMP-TA-01 to AMP-TA-03, deal with the identification of TLAAs that need to be updated for 60 years of plant lifetime by means of screening the Krsko CLB and the EQ program.

The AMP-TA-04 to AMP-TA-12 reports discuss and resolve TLAAs, resulting from the identification task. The AMP-TA-04 to AMP-TA-06 reports deal with metal fatigue. The AMP-TA-04 addresses the impact of the life extension to 60 years on the design of Class 1 RCS mechanical components supported by fatigue analyses. The AMP-TA-05 addresses the impact of the life extension to 60 years on the design of Class 1 auxiliary line components subject to fatigue analysis. The AMP-TA-06 addresses the potential effects of the reactor water environment on RCS components fatigue life during the period of extended operation.

The AMP-TA-07 and AMP-TA-08 reports discuss the reactor vessel embrittlement for a 60 years plant lifetime. The AMP-TA-09 report addresses the impact of cast stainless steel and stainless steel welds thermal aging for the extended period of operation. The AMP-TA-10 report considers updates in USAR Chapters 11 – Radioactive Waste Management and 15 – Accident Analysis [10]. The AMP-TA-11 report discusses TLAA analysis of Class 2/Class 3 primary sampling system lines fatigue and AMP-TA-12 provides the TLAA analysis for containment penetrations fatigue.

The fatigue results have been validated as acceptable for the extended period of operation. Fatigue monitoring is performed according to the transients' cycles counting method. After each fuel cycle the cycle counting is updated. The last available transient count relevant for the evaluation of mechanical fatigue of Class 1 components is documented in [47], Table 4-1A, B (Normal, Upset), and covers the data until the end of Cycle 28 (Autumn 2016). Current number of occurrences and also a bilinear extrapolation of the current number of occurrences of Normal, Upset and Test Condition transients to 60-year EOL is performed and the extrapolated values are also below the TS limit ([48], Table 4.7-1).

f) <u>How current "*state-of-art*", including R&D results, is taken into account</u>

NEK is the plant that uses NUREG-1801, Generic Aging Lessons Learned (GALL) report as guidance and takes into account the current »state-of-art« R&D results indirectly with the new revision of the GALL report. The NEK Aging Management Program was made in the beginning of 2010 based on revision 1 of GALL. At the end of 2010 revision 2 was issued. NEK AMP was evaluated against revision 2 and updated accordingly. This is the way of using R&D results in aging management that were accepted by the NRC. Independent Safety Evaluation Group (INSEG) has the responsibility to find all relevant worldwide R&D issues and put them into CAP program. Relevant R&D issues are also provided to NEK from SNSA.

g) <u>Consideration within the overall aging management program of modifications in the current licensing or regulatory framework</u>

NEK licensing department follows the current valid Slovenian and U.S. NRC regulation framework. Any change in regulation requirements applicable for NEK is identified and included into Corrective Action Program database. Additionally, NEK performs a 5-year AMP review according to MD-5 program and the regulation change would be identified as a finding. Based on such finding the corrective action is opened within the Corrective Action Program database, where all aspects of the regulation change are evaluated and appropriate solutions found. During the ordinary revision of any AMP the responsible engineer must review all reference documents for validity. If a change of regulation is found the program is revised with the valid regulation document referenced. Generally, the aging mechanisms are not very fast, allowing sufficient time for the implementation of any regulation modification.

h) <u>Identification of a need for further R&D</u>

NEK is the plant that uses NUREG-1801, Generic Aging Lessons Learned (GALL) report as guidance and takes into account the current »*state-of-art*« R&D results indirectly with the current revision of the GALL report [18]. NEK is a single reactor site plant, with Westinghouse NSSS, and alone it is not able to conduct its own extensive R&D as required for the purpose of aging management program development. As it is the US designed plant with US origin equipment and components, it is considered that the GALL report represents the closest industry operating experience in the fields of aging management for equipment, components and structures, as well as design and applied standards for the NEK case.

Additional tasks according to Aging Management Technical Specification for the National Assessment Reports [49]:

a) Strategy for periodic review of the overall AMP including potential interface with periodic safety reviews

NEK performed the second PSR for freeze date end of 2010 and the third is planned for the end of 2020, close to the end of the design life in 2023. An important part of the PSR is the NEK AMP review. The third PSR will have a special focus on AMP as it will be the last one before the transition into extended life. Independently of PSR the internal 5-year review of AMP according to MD-5 program will also be performed in 2020.

According to the overall AMP program MD-5 [1], NEK has to perform a periodic AMP review every 5 years. The product of the review is "*AMP Review Summary Report*", written as a plant technical report document, focusing on the following topics:

- Review of plant modifications performed in the last 5 years, their implementation, EAM-MECL changes, plant procedure changes etc. A list of modifications with designator (AMP=YES) must be added with brief conclusions about the proper implementation into EAM-MECL. For all NEW and MODIFIED components that potentially impact the AMP, the aging management review must be performed. The plant AMP programs and procedures must be reviewed accordingly. A list of affected systems (M), commodity groups (E, CS) must be developed;
- Review of industry practices, licensing requirements and new issues related to aging management of NPP. This requires a review of all NRC, EPRI, NEI and other AMP related documents;
- Review of plant operating experience (OE) in the last 5 years related to aging. This review must involve all evaluated OE in the last 5 years through the CAP program with evaluation and potential update of NEK programs and implementing procedures related to AMP;
- The conclusion of a review document must propose plant program changes and procedure revisions that are credited and used as AMP.

b) Incorporation of unexpected or new issues into the AMP

NEK is a GALL plant and so far NEK has not found unexpected or new issues, which are not addressed in NUREG-1801, revision 2 [18].

c) Use of results from monitoring, testing, sampling and inspection activities to review the overall AMP

Monitoring, testing, sampling or inspection activities performed by the aging management program are conducted according to the approved work order. The results of monitoring, testing, sampling and inspection activities are findings which can be: NI – no indication, NRI – non-relevant indication or RI – relevant indication.

The majority of results are NI and no further action is required. There are a few NRIs, for some of them a corrective action is initiated for refurbishment. Very rarely the RI is found with an aging mechanism, and an AMP repair action is initiated in NEK Corrective Action Program System.

In case the results of monitoring, testing, sampling or inspection activities are not bounded with the specific AMP then that specific AMP shall be reviewed to incorporate the new plant specific operating experience. The overall AMP will be also reviewed as the specific AMP is part of the overall AMP.

NEK has in place a Corrective Actions Program, quality assurance (QA) procedures, site review and approval process, and administrative controls are implemented in accordance with 10 CFR Part 50, Appendix B [36].

d) Periodic evaluation and measurement of the effectiveness of aging management

The first PSR in NEK was performed after approximately twenty years of plant operation. In the course of the first PSR a comprehensive and documented review was performed of the plant operational and design status. The review confirmed that the plant was as safe as originally intended and did not reveal any major safety issue. As a result it was found that NEK could safely operate, as a minimum up to the completion of the next Periodic Safety Review.

For the second PSR the Slovenian national legislation "Ionizing Radiation Protection and Nuclear Safety Act"[5] as well as regulations and directives issued by the Slovenian Nuclear Safety Administration (SNSA) were all observed. The implementation of the second PSR is also, in accordance with the current international safety policy and the IAEA Convention on Nuclear Safety. During the second PSR the complete AMP review was performed. The results of the review were provided in a report PSR2-NEK-1.4 [50]. Based on PSR2-NEK-1.4 five reported deficiencies were put into the Corrective Action Program database as part of PSR2 action plan. All of them were implemented during 2014.

During the period between two PSRs, changes occurred in safety standards, technology, analytical techniques; in NEK and regulatory body staffing, management structures and procedures; in environment; and changes were introduced due to plant modifications and aging. Aforementioned changes and their influences were reviewed and evaluated in order to determine their safety impact on NEK.

NPP Krško in its overall Aging Management Program (MD-5) requires that a thorough review of the Aging Management Program shall be conducted every 5 years besides the PSR. The result of the review is a list of findings, which are included as corrective actions into the NPP Krško Corrective Action Program.

2.5 Licensee's experience of application of the overall AMP

NEK started the implementation of the Aging Management Program with the issuance of the overall Aging Management Program, MD-5, rev. 0 with effective date: February 1, 2010, about 13 years before the end of 40-year design life (2023). Since then it has been supplemented and modified after each review.

All of the 36 AMPs that are part of NEK overall aging management program are currently implemented. On the annual basis NEK reports to the regulator, SNSA on the experience and results of the implementation of the AMP in accordance with JV9 [6].

NEK Aging Management Program has been reviewed according to the MD-5 requirement for 5-year periodic review and the work report ESD TR-19/15, "*NEK AMP Review Summary Report*" is issued. All recommended changes of processes, procedures and programs improve NEK AMP. They make AMP capable to properly identify every aging effect, analyze it in a proactive way and record it as an operating experience. Operating experience is recognized as a powerful tool because AMP is based on the operating experiences and only properly identified operating experiences will improve the future AMP.

The NEK AMP is consistent with NUREG-1801, Revision 2. The living AMP together with Maintenance Rule implementation process provide assurance for safe operation of the plant with its useful life extended from 40 to 60 years.

2.6 Regulatory oversight process

a) Assessment of the overall AMP and its modifications

In march 2009, the Krško NPP submitted the application for Ageing Management Program (AMP) approval and Safety Analysis Report change reviewed by technical support organization (TSO). The SNSA reviewed all licensing documentation, expert opinion report, Ageing Management Review (AMR) reports, Time Limited Ageing Analyses (TLAAs) and Programs/Procedures review and carried out a lot of walkdowns and meetings with the Krško NPP. When the review was completed and open issues (SNSA and TSO review) were resolved, the SNSA approved AMP itself and Safety analyses report (SAR) changes and licensing amendment was issued in June 2012. The approved AMP is one of the prerequisites for possible lifetime extension from 40 to 60 years of operation.

The SNSA supervises all AMP processes including AMPs implementation. Programs and procedures for ageing management implementation in general were proven to be appropriate and effective. The Krško NPP has no such experience that would require changes of methods, acceptance criteria or requirements in the programs.

Additionally, the Krško NPP has made some improvements to individual ageing management programs, based on the evaluation of changes derived from the 2. revision of GALL.

b) Inspection of implementation of the overall AMP

In 2016, one inspection was carried out on ageing management, where the SNSA covered all ageing management processes systematically and considered compliance with GALL rev. 2 implementation of individual programs and reporting of the Krško NPP at the system level. In the past, the SNSA carried out inspections, where topics from the current year's TPR, e.g. underground piping (part of concealed piping) and reactor vessel were discussed. A number of inspections at the level of individual systems, related to ageing management were also carried out each year, around 10 per year.

The SNSA also carried out an extensive inspection in 2017 regarding all relevant topics from TPR specifications. The main conclusions were the following:

- SNSA concluded that the Krško NPP has developed and implemented appropriate programs and procedures with methods for identification and monitoring the effects of ageing including the requirements for preventive and corrective measures implementation.
- The Krško NPP did not identify unacceptable degradations that would affect safe operation of the plant during the implementation of ageing management activities. All detected deviations were recorded within the corrective program implementation. Based on the analyses and applied standards and procedures provisions, appropriate preventive and corrective actions were carried out (e.g. ADP-1.1.133 »ASME Section XI, Repair / Replacement Program Implementation Procedure).
- Foreign operational experience connected with ageing management is considered systematically by the Krško NPP and appropriate preventive measures are implemented when necessary. An important and comprehensive preventive measure was the reactor vessel head replacement modification implemented at NPP Krško.

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

a) Regulator's assessment of the ageing management processes

The SNSA follows the whole AMP process in the Krško NPP by performing thematic inspections and oversight of the plant outages, through periodic safety reviews, modifications, review and assessment, operating experience feedback follow up, review and assessment of AMP documentation and other activities. The Krško NPP reports to the SNSA about the status of implementation of AMPs in the Annual Report. The basic program MD5, which covers all ageing management programs is reviewed by the Krško NPP at least every 5 years.

The Krško NPP programs and procedures for ageing management of electric cables, concealed piping, reactor pressure vessel and containment were reviewed in detail by extensive inspection in 2017. The bases for the review were TPR specifications and GALL requirements.

The regulator's opinion about the programs and procedures is that they are generally well prepared and are adequately applicable for the ageing management. The weaknesses of individual thematic fields of AMP, such as missing technical implementing procedures or non-implementation of certain activities from the programs were also identified by the regulator, and could be relatively easily eliminated. This will be done by the Krško NPP in the near future.

b) Regulator's experience from inspection and assessment as part of regulatory oversight;

During the thematic inspection, the SNSA had a detailed discussion with the Krško NPP regarding the assessment of ageing programs, implementation of systematic control, testing, sampling and inspection as well as implementation of preventive and corrective measures, and experience of the plant with specific programs for monitoring the ageing of safety related systems, structures and components (SSC). Specific experience facts especially from inspection carried out by the SNSA are written in the chapters regarding regulator's assessment and conclusions on ageing management of different SSCs covered by TPR specifications. In general, the Krško NPP developed and implemented appropriate programs and procedures for ageing management, and did not identify unacceptable degradations that would affect safe operation of the power plant.

c) Main strengths and weaknesses that have been identified either by the licensees or the regulator on the effectiveness of the overall AMP

Main strengths:

- Ageing management programs development is completed and proven to be in accordance with GALL requirements;
- Ageing managements programs Implementation phase is in a very comprehensive phase well before the end of the designed operational lifetime;
- Ownership responsibility for ageing management activities is on a very high level and personnel has generally a very proactive attitude.

Main weaknesses:

- Not all technical implementing procedures, which derive from ageing management programs, have been implemented yet.

d) The conclusion on the adequacy of the licensee's overall ageing management programme(s).

Aging management processes in the Krško NPP are generally well executed on a respective matter and comprise the requirements specified in NUREG-1801 - GALL, although some minor imperfections in the processes exist. The Krško NPP has well developed and implemented appropriate programs and procedures with methods for identification and monitoring of the effects of ageing and requirements for the implementation of preventive and corrective measures for aging management. By now the Krško NPP has not identified any unacceptable degradations that would affect safe operation of the plant in the course of the implementation of the aging control activities. All perceived deviations are recorded in the framework of the implementation of the Krško NPP corrective action program. Based on the analyses and specifications of applied standards and procedures, appropriate preventive and corrective actions have been carried out.

The Krško NPP should monitor and address foreign operational experience in the field of ageing and implement appropriate preventive measures also in the future. An important and comprehensive preventive measure in the past was to replace the reactor vessel head due to foreign experience with the primary water stress corrosion cracking (PWSCC), as well as for the minimization of the radiological burden on operator personnel and correlated economic reasons (reducing the scope of inspections) and the anticipated extension of service life of the component itself.

3 Electrical cables

3.1 Description of aging management programs for electrical cables

3.1.1 Scope of aging management for electrical cables

a) Methods and criteria used for selecting electrical cables

Scoping and screening for electrical cables for inclusion in NEK AMP was performed in accordance with NRC 10 CFR 54 [12]. The first step in the process was to determine which plant systems and structures are included within the scope of aging management. Systems, structures and components (SSCs) within the scope are those being in one of the categories, as required per [12]:

- safety-related (SR, 1E), or
- non-safety related (NSR, N1E), failure could adversely impact a safety related SSC, or
- relied upon in the plant evaluations to demonstrate compliance with regulations concerning either: fire protection (FP) [51], environmental qualification EQ [40], pressurized thermal shock (PTS) [52], anticipated accidents without scram ATWS [53] or station blackout (SBO) [54].

All plant systems, structures and components were reviewed against the above criteria and their intended functions defined and documented in relation with each criterion. The electrical cables are specifically listed in 10 CFR 54.21(a)(1)(i) as the components subject to aging management review since they are not periodically replaced.

For specific systems that are within the scope of aging management and contain at least one electrical component, a generic cable group (commodity) was assigned according to the nature of the circuit i.e. the purpose of its use: P for power cables, C for control cables, I for instrumentation cables and MV/HV- for medium/high voltage cables. For each commodity group insulation material was defined.

b) Processes/procedures for the identification of aging mechanisms

Detailed identification of the characteristics and routing (i.e. locations) related with each of the generic cable group/commodity (cables with similar materials, environment or operating conditions) was performed by undertaking a review of drawing information from the original (purchase) Technical Specification, Purchase Orders, Cable database (PCCKS) and other documentation.

The results were documented as follows:

- type of materials, and
- operating environment (divided among different buildings where each single cable or commodity group is routed).

The next important process was the identification of aging mechanisms. This was done in accordance with the GALL Report, NUREG-1801 [18]. GALL was checked weather a specific existing material /environment combination exists in NEK for every commodity group. As a starting point for "adverse" environment, the most severe environment that exists at the plant was used.

c) Grouping criteria for the Aging management purposes

For the evaluation of the aging degradation and aging mechanisms requiring management, the cables were grouped by the following designators: System, Material, Area (Buildings) and Nature of circuit (MV, P, C, I).
The majority of cables at NEK, regardless their function (SR or NSR) were, during the construction of the plant, purchased as the SR/1E nuclear-qualified cables. Exceptions are cables for systems outside the Nuclear island (Cooling towers system and Switchyard Equipment), which are all outside of scope of the AMP. NEK's Technical specifications for cables allow only two types of electrical insulations (EPR and

XLPE) and two types of cables jackets (fire retarding Hypalon - CSPE and Neoprene). This allowed a unified grouping by materials.

A large majority of the cables installed during the construction of the plant were Ethylene Propylene Rubber (EPR) insulated. This includes various types of EPR insulation from different manufacturers: Bonded EPR/CSPE (Boston Insulated Wire - BIW), High Temperature Kerite - HTK (Kerite) and OKOGUARD (Okonite). Cables have fire-retardant CSPE (Hypalon) jacket. Limited amount of cables were XLPE (Cross-Linked Polyethylene) insulated, with Neopren jacket, NIS cables had XLPE insulation and XLPE jacket (Rockbestos).

All cables purchased and installed within the scope of various safety related (SR) modifications were purchased as nuclear qualified 1E class cables with XLPE insulation and CSPE (Hypalon) jacket.

For grouping cables on the "Nature of a circuit" parameter, NEK complied with the NUREG-1801 [18] requirements, which recognizes 4 AMP categories:

- *XI.E1 Insulation Material for Electrical Cables and Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements*. This category includes all accessible power, instrument, control cables and connections;
- *XI.E2 Insulation Material for Electrical Cables and Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits*. This category includes electrical cables and connections used in circuits with sensitive, high voltage, low-level current signals, such as radiation monitoring and nuclear instrumentation;
- *XI.E3 Inaccessible Power Cables Not Subject To 10 CFR 50.49 Environmental Qualification Requirements*. This category includes all inaccessible or underground power cables (greater than or equal to 400 volts) exposed to adverse environments, primarily significant moisture;
- *XI.E6 Electrical Cable Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements***.** This category includes all cable connections associated with cables within the scope of license renewal that are external connections terminating at active or passive devices.

In those 4 AMP categories the following constituent elements are considered as the leading parts for the grouping and maintaining intended functions*: Conductor with connections, and Primary insulation.*

In accordance with EPRI TR 1003057 "LR Electrical Handbook", 2001 [55], the jacket is not considered a significant part of the cable from the perspective of the Aging Management Program. The cable jacket is designed for mechanical and fire protection of the cable insulation. The material of the jacket has therefore no significant effect on the aging process of the cable's primary insulation. The jacket properties are nevertheless used to determine the condition of the primary cable insulation. Since the CSPE and Neoprene have weaker properties than the main insulations (EPR, XLPE), the degradation of the jacket might indicate that there is an underlying degradation of primary insulation as well. In the TD-2D Cable Aging Management Program (CAMP) [56] visual inspection and Indenter Modulus testing of cable jackets is considered as the leading indicator of primary insulation condition. If any inacceptable conditions are found the cable has to be tested.

3.1.2 Aging assessment of electrical cables

Within NEK's AMP the aging assessment of Electrical cables is performed on the basis of the aging mechanisms defined in NUREG-1801 [18]. The assessment is implemented in accordance with the procedure TD-2D Cable Aging Management Program [56].

3.1.2.1 General assessment

The first step in the process of the general assessment was to define the commodity groups considering the function and the insulation material. Then the operating environment was defined, divided among different buildings where each single cable or commodity group is routed. All specific combinations (material-operating environment) were established and then compared with GALL [18] table VI A. If any potential aging mechanisms were found, then the relevant AMPs were selected and credited to ensure that necessary safety functions remain available throughout the extended lifetime.

For the Operating environment, the main stressors considered include exposures to significant moisture, temperature, radiation, mechanical wear as well as the exposure to chemicals. The Aging management programs in the GALL [18] applicable for the cables are: XI.E1, XI.E2, XI.E3, and XI.E6. These programs were determined through the identification of materials and operating environments for each commodity group.

- *Key standards and guidance used to prepare AMP for cable commodities include:*
  - NUREG-1801; Generic Aging Lessons Learned [18],
  - EPRI TR 1003057: LR Electrical Handbook [55],
  - EPRI-TR 1003317: Cable System Aging Management [57],
  - EPRI TR 103663: Integrated Cable System Aging Management Guidance [58],
  - EPRI TR 103834-P1-2: Effects of moisture on Life of Power Plant Cables [59],
  - EPRI TR 1016689: Medium Voltage Cable Aging Management Guide [60],
  - EPRI TR 109619: Guideline for the Management of Adverse Localized Equipment Environments [61],
  - EPRI TR 107514: Age-Related Degradation Inspection Method and Demonstration. [62].

- *Key design, manufacturing and operation documents used to prepare AMP for cable commodities include:*
  - Cable and raceway management database (PCCKS),
  - Cables purchase orders (PO) – original PO from the construction period documenting all design properties and materials,
  - Cable qualification reports – All cables from construction period (1E and N1E) are purchased as fire-retardant nuclear-qualified cables (meets requirements of IEEE-383 [63] and IEEE-323 [64] standards),
  - Technical specifications for cables – documents with defined requirements (construction details - thickness of insulation, materials to be used, required standards, QA requirements...), for all cable types (MV, Power, Control and Instrumentation),
  - Design environmental parameters specification – The design environment parameters requirements (from USAR [10]) were used as a starting point. All parameters higher than the following (i.e. (T>50°C, radiation >200 mS/h, 100% RH) are considered as "adverse",
  - Other cable related documentation – Cable Bill of Materials, Termination Sheets, Block Diagrams, Trays, Conduits, Underground routes, etc.

- *R&D programs:*
  - NEK prepared its own programs and implementation procedures with acceptance criteria for specific testing methods:
    - Indenter modulus (IM) acceptance criteria for thermally and radiation aged cables (HS-2011-02-T-IM [68]),
    - Line Impedance Resonance Analysis (LIRA) acceptance criteria (under development).
  - Cooperation with external research institutes and laboratories:
    - EPRI - Electrical Power Research Institute – Cable User Group meetings with operating and research experience exchange,
    - UJV Řež, CZ - Radiation monitoring in the Reactor Building cable areas for harsh environment determination with Gamma and Neutron detectors. Impact of different chemicals with oils on cable insulation is verified for oil resistance,
    - FKKT - University of Ljubljana, Faculty of Chemistry and Chemical Technology, Ljubljana, Slovenia, and FE - University of Maribor, Faculty of Energy Technology

– Determination of Differential Scanning Calorimetry (DSC) with acceptance criteria development for specific NEK materials,

- IJS – Institut Jožef Stefan - Radiation aging of cable insulation (gamma and neutron) and correlation of IM values with thermally aged cables. Preliminary results show that insulation materials age similarly from radiation as from temperature, and at low doses it is unlikely that radiation could cause severe deterioration of cable insulation.

➢ *Internal and external operating experience (OE):*
  - Systematic identification of external OE and evaluation of applicable OE (INPO, WANO, IRS, NRC…) is undertaken regularly and all the findings related with aging are recorded and addressed within the Corrective Action Program (CAP). If OE is recognized as applicable for NEK, an evaluation is undertaken and its results are incorporated in TD-2D program or procedures. NEK operating experience is shown that cables exposed to temperatures lower than 60°C do not experience significant aging that could cause cable failure. This is in correlation with 60-Year Service-Limiting Environments temperatures that are noted in chapter 3.2,
  - Regular participation and exchange of experience in International workshops (IAEA, EPRI – Cable Users Group, Training, Conferences, Meetings).

➢ *Establishment of acceptance criteria related to aging mechanisms.*
  - The acceptance criteria are established in accordance with GALL [18] and other applicable standards for any given inspection method as defined in the TD-2D [56] e.g.:
    - Adverse localized environmental conditions,
    - Visual inspection criteria and testing methods,
    - *Dissipation Factor Test (Tan δ),* in accordance with IEEE 400.2 standard [69],
    - Insulation resistance, per ANSI/NETA ATS-2009 [70],
    - Indenter modulus for NEK specific cable types, in accordance with HS-2011-02-T-IM [68].

3.1.2.2 Aging assessment of power cables subject to adverse environment

The electrical cables used at NEK grouped in the "Power Cables" category include inaccessible (e.g. in a conduit or direct buried) 6.3kV (referred as Medium voltage – MV) and 400V power cables (referred as Low Voltage – LV) that are exposed to significant moisture. GALL [18], defines that MV cables range from 2 kV to 35 kV, and Low Voltage cables are up to 2kV. Recognized aging mechanisms for this group (in accordance with GALL [18], Table VI A) are exposures to significant mechanical, moisture, temperature, radiation and chemical effects.

All Medium voltage cables in NEK (regardless on their safety function) and sample of underground LV power cables are within the scope of CAMP [56] and Preventive Maintenance (PM) program ADP-1.4.459 [71]. Those programs are consistent with GALL [18] XI.E1 Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements and XI.E3 Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management programs.

Cables are visually inspected, tested and all manholes are visually inspected on a monthly basis, including water pumping as needed, assuring that cables are not exposed to significant moisture.

3.1.2.3 Aging assessment of High Voltage Low Level Current instrumentation cables

Recognized aging mechanisms for this group (in accordance with GALL [18], Table VI A) are exposures to significant mechanical, moisture, temperature, radiation and chemical effects.

In NEK, a limited number of high voltage low current cables are installed. These are nuclear instrumentation cables (NIS) and radiation monitoring cables (RM). All circuits and channels surveillance / calibration results are evaluated in accordance with I&C specific programs. The procedures that are consistent with GALL [18] XI.E2 Insulation Material for Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits aging management program.

Additionally, a sample of High Voltage Low Level Current instrumentation cables are listed in PM program ADP-1.4.459 [71], consistent with GALL [18] XI.E2 and are visually inspected and tested (insulation resistance and LIRA).

3.1.2.4 Aging assessment of high voltage cable

The only high voltage cable within the scope of AMP is 110 kV connection to outside power supply, used as the alternative external power supply required by NEK's Technical Specifications and credited in the plant specific Station Blackout analysis. Initially installed Oil-filled cable was replaced with a new XLPE insulated cable. New XLPE cable design prevents water intrusion into the primary insulation with longitudinal Al tape and watertight overlapping tape. Since it is a new cable designed for underground installation and direct burial no significant degradation is expected. Due to its importance, high voltage cable is periodically tested within the PM program for high voltage equipment - ADP-1.4.456 [72].

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

3.1.3.1 General activities

Environmental parameters (temperature, radiation, moisture, vibrations, etc.) in cable spreading area are monitored and visually inspected every 5 years. The inspection includes all underground cable routes also addressing structural deficiencies (cracking of concrete, cable tray corrosion, modification of cable trays).

Visual inspection of underground manholes is conducted monthly and after heavy rain when water pumping is required, assuring that cables are not submerged.

Cables in the adverse environment are sampled in accordance with EPRI TR-107514 [62]. All inaccessible MV cables and sample of LV Power, Control and Instrument cables are added in PM program ADP-1.4.459 [71].

A visual inspection of the whole route of each cable (within the scope of the AMP) is conducted during testing, which includes:

a) Testing of all cables within the CAMP TD-2D [56] and the PM procedure ADP-1.4.456 [72]:
    - Insulation resistance (IR)- acceptance criteria in accordance with ANSI/NETA ATS-2009 [70],
    - Line resistance of cable with connections (continuity) – acceptance criteria in accordance with electrical properties, cross section and length of cable,
    - Infrared Thermography inspection of power and control connections, acceptance criteria are in accordance with ASNT Recommended Practice No. SNT-TC-1A [73] and PDM-4.200 [74].

b) Testing methods used when required:
    - Indenter Modulus for cables in adverse environment (temperature and radiation) – acceptance criteria are in accordance with HS-2011-02-T-IM [68],
    - AC Dielectric Withstand test (HI POT) of power cables with unacceptable testing results of other methods (insulation resistance, dissipation factor, etc.).

c) Other methods used for additional evaluations:
    - Line Impedance Resonance Analysis (LIRA) on wide frequency range - acceptance criteria are predefined with instrument and plant specific acceptance criteria is under development,
    - Chemical laboratory Analysis (DSC, FTIR, etc.), acceptance criteria are predefined with standards and plant specific acceptance criteria is under development.

3.1.3.2 Medium voltage cables subject to adverse environment

Specific activities for MV (6.3 kV) cables within the scope of PM program ADP-1.4.459 [71] include testing with frequency of 6 years:

- Dissipation Factor Test – TD (Tan δ / Loss Angle Test) with two testing methods using different frequencies (0,1 Hz - Very Low Frequency and 50 Hz – Power Frequency). Acceptance criteria for VLF are in accordance with IEEE Std 400.2 [69],
- Partial Discharge - PD with two testing methods (0,1 Hz - Very Low Frequency, and 50 Hz – Power Frequency) is used only for additional evaluation. The acceptance criteria for VLF are in accordance with IEEE Std 400.3 [75].

3.1.3.3 Neutron flux instrumentation cables subject to adverse environment

High voltage low current cables are nuclear instrumentation (NIS) cables and radiation monitoring (RM) cables. All circuits and channels surveillance and calibration results are evaluated in accordance with specific I&C procedures. Sampled cables are listed in ADP-1.4.459 [71], visually inspected and tested (insulation resistance and LIRA).

3.1.3.4 The High voltage cable

High voltage (110kV) cable is periodically tested within the PM program for high voltage equipment - ADP-1.4.456 [72].

3.1.3.5 Key features of the inspection program TD-2D

The main features of TD-2D, Cable Aging Management Program (CAMP) established in (or implemented since) 2010 include:

a) Visual inspection of cable spreading areas to identify local adverse environment:
   - Measurement of parameters (temperature, radiation, moisture, vibration, etc.) in building with identification of local adverse ambient – "Hot Spot". All cable spreading areas in buildings are inspected every 5 years with thermal camera for searching "Hot Spots". In more than 30 locations in containment, temperatures are measured, and collected with data loggers during two operating cycles (36 months). On 50 locations, the gamma dose is measured with alanine pellets (in each 1 operating cycle - 18 months).

b) Visual inspections of sampled accessible cables for aging effects:
   - Accessible electrical cables and connections are visually inspected for cable jacket and connection insulation surface anomalies, such as embrittlement, discoloration, cracking, melting, swelling or surface contamination. Inspection of cable jacket and connection insulation surfaces is used to confirm that insulation is capable to perform its intended functions. Visual inspection of specific cable is conducted at least once every 10 years.

c) Inspection for water collection in cable manholes:
   - All manholes are monthly visually inspected and water is pumped, if needed assuring that cables are not submerged.

d) Electrical tests of cables in adverse environment are conducted for each cable with identified degradation mechanism.

e) Estimating expected life time for exposed cables is done using tests, calculations or laboratory investigation to predict residual life of cable insulation. Aging Management Software Support Platform (COMSY) is used to evaluate all environmental data, cable measurements and to calculate residual lifetime of cable.

f) Corrective actions to repair and replace damaged cables or to eliminate and prevent aging effects on cables or connections in the future.

3.1.4 Preventive and remedial actions for all electrical cables

Preventive and remedial actions are applicable to all groups of cables: power cables subject to adverse environment (LV and MV), high voltage low current cables (NIS, RM) and high voltage cables. PM program ADP-1.4.459 [71] is implemented with defined scope activities and frequency as explained in chapter 3.1.3.

When an adverse localized thermal environment is identified, a corrective action is required to be opened in the Corrective Action Program (CAP) in accordance with ADP-1.0.020 [38] *Corrective Action Program* procedure. The first preventive step is replacing, repairing or upgrading of thermal insulation that is subjected to a source of heat and radiation. The next action, for accessible cables, is the indenter modulus (IM) testing or other electrical testing. If severe thermal aging of the cable insulation is identified, removal, replacement and rerouting of the affected cables (or sections) are conducted.

If the temperature-adverse environment criteria for severe cable degradation is not reached, but the temperature is still higher than the maximum design plant temperature, periodic assessment of the cable insulation condition is undertaken to verify the rate and severity of cable degradation. Corrective action may be taken at the appropriate time if degradation is determined.

All severe thermal "Hot Spots" (with temperatures higher than 60°C) identified during the assessment campaign (periodical visual inspections of all buildings) were subsequently eliminated. Cables that remain exposed to elevated temperature (i.e. higher than the plant's design temperatures), but not high enough to cause the deterioration of the cable insulation are within the scope of the TD-2D program.

The acceptance criteria for testing of cables are categorized in 3 groups: "*Good*", "*Study*" and "*Action required*". If the result of testing is not within the "*Good*" range, it is introduced in the Corrective Action Program. If it falls into the "*Study*" range, the frequency of periodic inspection is increased depending on testing results and engineering judgment. If the measurements results show as being in the "*Action required*" range, immediate actions are taken including as appropriate additional testing, repairing, rerouting or replacing a cable.

During the initial visual inspection of underground cable routes, all manholes were visually inspected and deficiencies eliminated: cables routed on the floor of underground channels were lifted onto cable trays and an automatic water pump for drainage was installed (NSR-N1E cables). Underground cable conduits with SR cables are visually inspected with borescope to ensure that water is not trapped somewhere in the raceways.

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

The discussion below is applicable to all groups of cables: power cables subject to adverse environment (LV and MV), high voltage low current cables (NIS, RM) and high voltage cables.

Adverse environment detection

All cables in NEK regardless of their function or application (SR or NSR) are purchased as the 1E nuclear-qualified cables in accordance with the applicable technical specifications. Technical specifications allow two types of electrical insulation (EPR and XLPE) and 2 types of cables jacket (fire retarding Hypalon-CSPE and Neoprene).

In EPRI FR 1003057: *LR Electrical Handbook* [55] document the following 60-Year Service-Limiting Environments temperatures are stated for these materials:

- Insulation material
  - EPR: 75.0°C
  - XLPE: 86.6°C
  - HTK: 85.2°C

- Jacket material:
  - CSPE: 75.0°C
  - Neopren: 41.7°C

As the maximum plant design temperature is 50°C, it is clear that cable insulation has a large temperature margin and that only neoprene jackets could be degraded. Since those cables have XLPE primary insulation, even with degraded jacket, the cable intended function is not expected to be compromised. Area temperature measurements undertaken during periodical visual inspections of all buildings indicate that actual temperatures are typically much lower than the design temperatures.

During the visual inspections of buildings, about 70 potential localized temperature "*Hot Spots*" were found.

In about one third of those locations, approximately 60 cables were rerouted and 20 were replaced due to jacket damage/cracks. The primary insulation of replaced cables was still in acceptable condition - undamaged.

In about one third of locations cable jackets were undamaged and thermal insulation of piping was upgraded and/or the hot spot eliminated.

Remaining locations with cables exposed to elevated but not severe temperatures ($50 < T < 60$ °C) are within the scope of ADP-1.4.459 [71] program. Those are being monitored/tested with defined activities (visual inspections and electrical testing) and frequencies as explained in 3.1.3.

The Indenter modulus (IM) testing of CSPE cable jacket is used for cable insulation evaluation. IM values for new cable jacket are in the range of 9-12 N/mm. The overall IM jacket results in general environment are in the range of 13 to 17 N/mm. Cables with IM of jacket above 30 N/mm are added to the scope of ADP-1.4.459 for trending as the jacket cracking starts at IM higher than 70 N/mm. Although the primary insulation would still remain undamaged, this is the point when the cable replacement becomes relevant. Cracking of the primary insulation (EPR, XLPE) is expected to occur at the IM values of jacket higher than 200 N/mm.

Radiation measurements determined that only a limited number of cables are exposed to higher radiation doses. According to cable qualifications and NEK own research, highest doses could cause detectable effects on cable insulation though not severe degradation. All cables are qualified to 200 Mrad dose. Combined (neutron and gamma) cumulative dose for 60 years on location with highest measured radiation is less than 80 Mrad. Additionally, NEK researches (indenter measurements of irradiated cables) showed that the cable insulation receiving a dose of 200 Mrad still remains in the "trending/study" range (IM values are below 70 N/mm). Cables irradiated with 100 Mrad have IM values from 18 to 24 N/mm, which is well below the "trending" range.

MV cables measurements and activities

NEK concluded the first cycle aging management inspections and testing for all MV cables with the dissipation factor (Tan δ - VLF / 50 Hz). Results of most Tan δ measurements are in the "*Good*" range. Four out of 81 MV cables are in "*Study*" range. Those 4 cables are all N1E underground cables with splices. Slightly elevated measurement results (between "*Good*" and "*Study*" range) are generally expected when splices are present. Nevertheless, for those 4 cables the frequency of inspections is reduced to 3 years.
A monthly inspection of manholes is conducted to assure that cables are not submerged. Survey is implemented to measure humidity in underground manholes.
Spare MV cables and splices have been purchased to assure that any cable with unacceptable testing results can be replaced/repaired immediately.

Software platform

Software support (COMSY) for calculation of residual life time (temperature, radiation), risk ranking (electrical, mechanical, chemical effects) and reporting is developed in cooperation with AREVA.

All cables located within defined areas (buildings) and trays, where the cables are routed and all relevant cable data (material, year of installation, inspections, etc.) are entered into the COMSY software.

For all areas and trays, the area parameters (temperature, radiation doses and humidity) are defined. Temperature and radiation parameters are used to calculate residual life time. For temperature-related residual life time, Arrhenius equation is used. For radiation damage, the Power low (calculation of residual life time based on original qualification doses and actual doses cables are exposed to) is used. It is possible to store measurement results for every cable. Based on the defined acceptance criteria software calculates and display risk-ranking matrix. Additionally, software recommends the next inspection date and also allows to graphically trend the results of measurements.

There is also a feature in the software to add "*Hot Spot*" areas (with elevated environmental parameters). Then the software calculates the new residual lifetime, risk ranking and recommends the next inspection date taking those parameters into the account.

Program Health Report
Program Health Report for CAMP is established as a useful tool to display the status of program performance. It has four cornerstones with defined sub-indicators:

- Personnel: Qualification and Experience, Bench Strength, Technical Team Participation, Industry Participation, Supervisor Oversight,
- Infrastructure: Database Health, Strategic Plan, Corrective Action Completion, Self-Assessment,
- Implementation: Program Critical Condition Reports, Work Management Performance, Manholes Inspections- Cable Submergence, Tan Delta Cable Testing,
- Equipment: In – Service Cable Failures, Water Pumping from MHE, Medium Voltage Cable Health.

Relative Value and *Quality Points* performance indicators are calculated and color-coded (red <50%, yellow <68%, white <83% and green >83%) for every cornerstone.
The Health Report is produced quarterly and graphically displayed for easy demonstration of the program status in the Summary Program Health. It helps the management to monitor the program implementation activities and to prioritize important topics. If any of the performance indicators value drops to an unacceptable level, action is initiated. Current status of overall indicator is acceptable - white (75% of maximum performance). The Goal line is higher than 70% of maximum performance and the program is aiming to all "green" performance.

CAMP activities under other electrical and instrumentation preventive maintenance (PM) programs
As an addition to TD-2D, all PM procedures of electrical and instrumentation equipment are updated to be provided for inspection of aging of passive components connected to active components: cables and all types of electrical connections including grounding connections, terminal blocks, connectors and splices.

The polymer insulation of electrical cables and connections (part of cable/connection within electrical component) is, per all electrical and instrumentation PM procedures, visually inspected for surface anomalies such as embrittlement, discoloration, cracking, swelling, melting or surface contamination. The accessible cables and connections must be free from unacceptable visual indications or surface anomalies, which would suggest that the conductor insulation degradation might be occurring. All types of connections are being inspected for corrosion, oxidation and loosening of bolted connections due to thermal cycling, ohmic heating, electrical transients, vibration and chemical contamination.

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

a) Assessment of the ageing management processes

Ageing management of electrical cables is covered under the Cable Aging Management Program TD-2D (CAMP)issued in 2010. The purpose of CAMP is to provide assurance of functionality for electrical cables and connections exposed to ageing effect because of adverse environments caused by heat, radiation, water, chemical or mechanical wear. Identification of potential adverse localized environments or adverse service conditions and management of cable insulation and connections, are the main concern of this program. The program is based on NUREG – 1801 (GALL), Maintenance Rule (10 CFR 50.65), License renewal Rule (10 CFR 54) and other EPRI and INPO technical reports.

CAMP could be divided on more parts:

- List of cables in adverse environment;
- Programs and procedures in connection with CAMP program (e.g. EQ, Corrective action program…);
- Cables maintenance and implementation procedures;
- Cables prediction tool (COMSY);
- Corrective actions, testing, monitoring and trending.

The general SNSA opinion is that CAMP is quite well developed, but also some weaknesses are identified.

b) Experience from inspection and assessment as part of its regulatory oversight

The SNSA obtains information about the status of CAMP implementation with two approaches:

- Regular reporting of the Krško NPP
- Thematic inspections performed by SNSA.

The Krško NPP should report annually and quarterly to the SNSA. Part of these reports is dedicated to the reporting about aging management. The most important is summary of activities performed within aging management programs and operational experiences regarding aging.

The SNSA performs one thematic inspection per year with the purpose to assess the status of implementation of CAMP program.

Insights and findings from the 2015 inspection are the following:

- 55 activities were performed on cables and connections (visual inspection of cables were performed in reactor building, auxiliary building, turbine building), testing and diagnostic for LV and MV cables;
- From 2013 to 2015 measuring neutron and gamma radiation on 50 locations and temperature measuring on 14 locations in reactor building was performed;
- 17 issues were identified, especially because of new »hot spot«;
- General problem was with water in manholes and submergence of cables;
- The Krško NPP introduce new testing method LIRA.

Insights and findings from the 2016 inspection are the following:

- CAMP program was revised (rev.2), because of additional requirements of NUREG 1801 - GALL rev. 2 and the list of cables in adverse environments was revised;
- New preventive maintenance procedure for cables was developed - ADP-1.4.459;
- COMSY-CABLE software (estimate the remaining life of passive cables and connectors) was put in place;
- 50 activities were performed on cables (visual inspection of the cables, testing and diagnostic for LV and MV);
- 13 issues were identified, especially because of new »hot spot«.

Insights and findings from the 2017 inspection:

- Overall assessment of CAMP program was performed. The main strengths and weaknesses are described below.

c) The main strengths and weaknesses that have been identified either by the licensees or the regulator on the effectiveness of the SSC specific AMPs.

Main strengths:

- Personnel responsible for cable preventive maintenance have very positive attitude regarding cable maintenance (e.g. good safety culture, a lot of technical knowledge, etc.).

Main weaknesses:

- The procedure for diagnostic testing of electrical cables is still in draft version and will be prepared in 2018.

  Maintenance procedure describes the sampling of cables. The Krško NPP used criteria from technical report EPRI TR-107514, Chapter 4 (Sampling program description). This criterion is applicable in case when cables are not degraded. Criteria for sample selection in case of degradation finding will be revised in 2018 and if necessary the list of sampled cables will be extended.

The Krško NPP is aware of the above mentioned noncompliances and they will be resolved in 2018. SNSA's opinion is that the above mentioned non-compliances are important for selection of sampling list of electrical cables and will follow the implementation of the Krško NPP actions for improvement.

Electrical cables and connections are very important for plant safety. In addition electrical cables distribution represents one of the biggest systems in the plant. The SNSA identified that insufficient number of plant staff work on CAMP program. SNSA's opinion is that more staff is needed for activities regarding aging management of cables for future.

d) The conclusions on the adequacy of the licensee's SSC specific ageing management programme(s)

Ageing management of electrical cables is covered under the Cable Aging Management Program (CAMP-TD-2D). The program is in compliance with NUREG – 1801 (GALL), the Maintenance Rule (10 CFR 50.65), the License Renewal Rule (10 CFR 54) and other EPRI and INPO technical reports and guidelines. A general SNSA opinion is that CAMP program is quite well developed, but it is important to resolve all open weaknesses and discrepancies identified.

4 Concealed pipework

4.1 Description of aging management programs for concealed pipework

4.1.1 Scope of aging management for concealed pipework

NEK had started monitoring the concealed pipework with inspections of buried portion of Essential Service Water system even before the Aging Management Program was established. The main trigger to initiate the visual and volumetric inspections was the US NRC Generic Letter 89-13 – "*Service Water System Problems Affecting Safety-Related Equipment*" [76]. The Generic Letter requested nuclear power plants to establish a routine inspection and maintenance program for the open-cycle service water system piping and components, so that corrosion, erosion, protective coating failure, silting and biofouling could not degrade the performance of safety-related systems supplied by the essential service water. In 2003, NEK performed the first partial volumetric inspection of the Essential Service Water system buried piping in order to detect corrosion degradation of pipe wall, using long range-guided ultrasonic waves.

NEK Aging Management Program was developed in 2010. Reflecting the outcome of the scoping process in accordance with NRC 10 CFR 54, "*Requirements for Renewal of Operating Licenses for Nuclear Power Plants*" [12] and NEI 95-10, "*Industry Guideline for Implementing the Requirements of 10 CRF Part 54 – The License Renewal Rule*", [16], the following systems were included in the Aging Management Program for concealed components: Essential Service Water System, Diesel Generator Oil System and Fire Protection System. These systems were selected due to their importance to plant safety and since it was recognized that they encompass components (pipework, tanks) that are buried underground in conditions that could potentially lead to corrosion. Therefore, the components have to be monitored under the AMP.

The purpose of the aging management program for concealed components is to implement surveillance activities for buried and underground piping and tanks. Underground piping and tanks surveillance program was developed in accordance with guidelines provided in NUREG-1801, Generic Aging Lessons Learned (GALL) Report [18]. Specific analyses which were the basis for the NEK AMP were developed during the Aging Management Review (AMR) phase. In connection with NEK-ESD TR-05/14, »*Aging Management Program Evaluation Report*« [19], NEK modified TD-2Z, "*Buried and Underground Piping and Tanks" program* [77] that implements specific aging management recommendations.

The Program includes surveillance and preventive measures to mitigate corrosion by protecting the external surface of buried and underground steel piping and tanks. Surveillance and preventive measures are in accordance with standard industry practice based on National Association of Corrosion Engineers (NACE) Standards.

The program relies on preventive measures, such as coating, wrapping, cathodic protection and surveillance based on NACE Standard RP-0285-2002 [78], NACE Standard SP-0169-2007 [79] and NACE Standard RP-502-2008 [80] to manage the effects of corrosion and to assure intended function of buried tanks and piping, respectively.

The effectiveness of coatings and cathodic protection system, per standard industry practice, is determined by measuring coating conductance by surveying pipe-to-soil potential and by conducting the bell hole examinations to visually examine the condition of coating.
Inspections of damaged coatings and degraded material condition are conducted periodically. At least the visual inspection is performed. Furthermore, non-destructive testing such as ultrasonic testing and eddy current testing are effective methods to measure surface condition and the extent of wall thinning of the piping. Inspection is performed every time, when for whatever reason a component becomes accessible.

Inspection locations are selected based on susceptibility to degradation and consequences of failure. For directed inspection, the visual method is supplemented with surface and/or volumetric non-destructive testing if significant indications are observed.

In accordance with accepted industry practice NACE Standard RP-0285-2002 [78] and NACE Standard RP-0169-2007 [79] the assessment of the condition of coating and cathodic protection system is conducted on a regular basis and compared with predetermined acceptance criteria.
When tanks are emptied and cleaned, bottom and side wall ultrasonic thickness measurement or magnetic flux leakage method is performed in accordance with TD-2Z, "*Buried and Underground Piping and Tanks*" program [77].

The directed inspections for buried tanks, as required by NUREG-1801, Rev. 2, XI.M41 Table 4c [18] are conducted from external surface of the tank using visual techniques or from the internal surface using volumetric techniques (UT thickness measurement). If a tank is inspected from the external surface, a minimum of 25 % coverage is required and the inspected area shall include at least some areas of both the top and bottom of the tank. If a tank is inspected internally by UT, at least one measurement per square foot of thank surface is required in accordance with [18].

NEK Emergency Diesel Generator tanks DO100TNK-001 and 002 are Code Class 3. Therefore, per Ref. [18] only one inspection should be conducted during each 10-year period beginning 10 years prior the entry into the period of extended operation.

a) Methods and criteria used for selecting concealed pipework within the scope of aging management

Methods and criteria used for selecting concealed pipework within the scope of aging management are defined in accordance with NUREG-1801, Rev. 2, "*Generic Aging Lessons Learned (GALL) Report*" [18], Chapter and include:

- V. Engineered Safety Features:
  - V.D1 - Emergency Core Cooling System (Pressurized Water Reactors);
  - V.E - External Surfaces of Components and Miscellaneous Bolting.
- VII. Auxiliary Systems:
  - VII.C1 - Open-Cycle Cooling Water System (Service Water System);
  - VII.C3 – Ultimate heat sink;
  - VII.G – Fire Protection;
  - VII. H1 - Diesel Fuel Oil System;
  - VII.I - External Surfaces of Components and Miscellaneous Bolting.
- VIII. Steam and Power Conversion System
  - VIII.E – Condensate system;
  - VIII-G – Auxiliary Feedwater System;
  - VIII-H – External Surfaces of Components and Miscellaneous Bolting.

Based on flow diagrams of mechanical systems and walk-downs, all buried and underground pipelines and tanks at NEK were identified. NEK's Buried and underground piping and tanks are only the following ones:

- VII.C1 - Open-Cycle Cooling Water System (Service Water System);
- VII. H1 - Diesel Fuel Oil System;
- VII.G – Fire Protection.

NEK buried and underground piping and tanks program includes each underground pipeline or tank that is classified as code class/safety-related or contains hazardous materials and is constructed from a material susceptible to degradation. All piping and tanks fulfilling those criteria have to be appropriately examined under the NEK's AMP.

b) Processes/procedures for the identification of aging mechanisms related to concealed pipework

For detection of surface corrosion and coating protection degradation as well as for the identification of any additional aging degradation mechanisms, NEK uses visual examination procedures. For the volumetric examination NEK employs volumetric methods like ultrasonic examination (conventional and long range guided waves) and magnetic flux leakage examination.

Additionally, according to the requirements of NEK's Aging Management Program, Essential Service Water system is subject to the requirements of the ASME Section XI, IWA-5000, System Pressure Test [81] and is tested in accordance with NEK ISI Program TD-2E/4 [82].

c) Grouping criteria for aging management purposes

As NEK does not have too many concealed pipework therefore, there is no need for grouping. All the concealed pipework is considered of high importance. The following systems were selected for Buried and Underground Piping and Tank Surveillance Program as defined by the Aging Management Review:

- Diesel Oil Storage System;
- Essential Service Water System;
- Fire Protection System (only buried piping portions made of steel).

The following table specifies affected components, materials and environment:

SYS	SSC Tag#	Type	Description/Material /Thickness	AMR Group – Internal Effects	AMR Group – External Effects
DO	DO100TNK-001	TNK	DIESEL FUEL OIL STORAGE TANK 1/ CS / 12.0 mm	CS in fuel oil	CS in underground
DO	DO100TNK-002	TNK	DIESEL FUEL OIL STORAGE TANK 2/ CS / 12.0 mm	CS in fuel oil	CS in underground
DO DO	DO-YRD-PIP3 DO-YRD-PIP2	PIP PIP	PROCCES PIPES/ CS/ 6.0 mm PROCCES PIPES/CS/4.0 mm	CS in fuel oil CS in fuel oil	CS in underground CS in underground
SW	SW-YRD-PIP7	PIP	PROCCES PIPES/CS/ 9.53 mm	CS in river water	CS in underground
FP FP	FP-YRD-PIP9 FP-YRD-PIP29	PIP PIP	PROCCES PIPES/CS/4.0 mm PROCCES PIPES/CS/4.0 mm	Galvanized steel in air CS in river water	GS in underground CS in underground

It is very important to point out that NEK has no concealed pipework containing radioactive effluents.

4.1.2 Aging assessment of concealed pipework

SP-G536A - Technical Specification Piping Line Specifications [83], NEK flow diagrams and as-built drawings were the main documents to identify and locate pipework that should be included in concealed pipework programs. Inspection procedures were developed by inspection companies and were reviewed and accepted by NEK.

As already mentioned in Chapter 2.4 h), NEK does not perform its own R&D programs but rather supports EPRI's UT Guided Waves R&D program by providing inspection results and transferring its own experience. At the same time, NEK actively participates in EPRI Buried Piping Committee that directs R&D in this field.

NEK follows worldwide operating experience related to the concealed pipework. Close contacts with EPRI NDE department and especially with experts dealing with concealed pipework issues is the most effective way to be informed about the latest industry events and experience. NEK participates in EPRI annual conference that also covers concealed piping issues. Collected information is discussed and used in evaluation under the NEK Corrective Action Program.

It is important to point out that NEK was one of the first nuclear power plants that included UT Guided Waves in inspection of concealed piping. The first inspection on SW piping was performed in 2003. By now, NEK has accumulated quite broad experience on this application.

The results of all inspections performed within the AMP confirmed very good condition of NEK's concealed pipework and tanks. No significant degradation was found. Therefore, neither the corrosion rates nor the remaining life of the components need to be calculated. Concealed pipework and tanks have been in operation for more than 35 years. Since they are properly protected and inspections so far have not revealed any significant degradation, it can be assumed that they will stay in this condition for next 25 years at least. In addition to good protection, Essential Service Water system has been protected by cathodic protection system as a continuous monitoring system (see section 4.1.3).

NEK will continue to inspect all concealed components mostly using directed inspections. All inspections performed on concealed components are credited as baseline inspections. It is NEK's intent to repeat the complete inspection program within the next 10 years. This will provide a reasonable assurance that the corrosion is neither progressing nor the structural and leakage fitness for service has been impaired.

In case of any degradation found, NEK will take appropriate actions, commensurate with a degree of degradation found. This will be based on a detailed assessment of degradation, using non-destructive methods to determine the remaining wall thickness and analytical evaluation and calculation of the minimum allowable wall thickness in accordance with ASME Section XI criteria. Based on this, trending, repair or replacement of degraded components, in accordance with ADP-1.1.133; ASME Section XI Repair/Replacement Program and Implementation [84] and ADP-1.0.020; Uporaba korektivnega programa [38] will be undertaken.

NEK follows industry and science research and development in the area of new methods, to increase its inspection capabilities. The goal is to achieve 100% inspections of total surfaces and lengths.

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

4.1.3.1 Methods and activities

Concealed pipework and tanks in NEK are monitored/inspected using several methods:

- Visual inspection (protective coating condition, component material condition, etc.);
- Ultrasonic or magnetic flux leakage;
- Monitoring of cathodic protection current (Essential Service Water System pipework);
- Monitoring of system leakage (Fire Protection System).

Visual inspection is always performed when concealed pipelines or tanks are excavated to perform maintenance, examinations or for any other reason. Pipeline protective coating is thoroughly inspected to determine the overall coating condition and to identify potential damages. An overall visual inspection for leakages is also performed. If possible, pipeline or tank condition is inspected for signs of any corrosion occurring. If any damage on the coating or material, leakage or other deviations are discovered by visual inspection, appropriate corrective actions are initiated under the NEK Corrective Action Program [38].

Ultrasonic or magnetic flux leakage methods are used as non-destructive volumetric methods for pipeline/tanks inspections. They are used to determine pipeline or tank wall thickness and the presence of corrosion.

Pipes are partially excavated, cleaned and examined using conventional ultrasonic examination or long range guided ultrasonic examination and especially for the Diesel Oil tanks – magnetic flux leakage examination.

Monitoring of cathodic protection current is regular practice at NEK. The method is based on monitoring coating conductance versus time or the current requirement versus time. Measured values are compared with predetermined parameters values, which provide an indication of the status of protective coating and

SW piping cathodic protection condition. NEK Essential Service Water system has been cathodically protected since 2010.

The requirements for performing these examinations are established in NEK procedure PME-4.300 [85]. The procedure provides the guidelines for continuous monitoring of cathodic protection system parameters (set to negative 850 mVDC). In case when measured parameters deviate from the predetermined values or there is an increase in the required current, this is an indication of degradation of protective coating and/or wrapping layer. As a consequence corrosion on the component material could occur. In this case, corrective actions are implemented in accordance with the requirements of NEK's Corrective Action Program [38]. A close interval pipe-to-soil potential survey is used to identify locations, where degradation occurred, and repair is needed. Until now, no such degradation has been found at NEK.

Fire protection system leakage is monitored by the frequency of starts of the jockey pump. The Jockey pump is maintaining pressure in the fire protection system at a predetermined value. When pressure in the system drops, the pump will automatically start and after reaching the set pressure, the pump will automatically stop. If significant leakage occurs in the system, the pressure will drop more rapidly and the frequency of starts will increase. By monitoring pump starts frequency, it is possible to detect leakages, which could be related with concealed FP pipework. In such a case, a detailed search for location of leakages will be initiated and, upon identification, repairs will be performed. The document LR-ISG-2015-01 [86] in Appendix B, item g.iii.(b) states that monitoring of the activity of the jockey pump is an acceptable alternative to the preventive actions for the Fire Protection system. NEK continuously monitor the activity of the jockey pump and will respond in accordance with NEK's Corrective Action Program [38], in case of increased number of pump starts and run time. Until now no leakages on the Fire Protection concealed pipework have been detected in NEK.

Inspection activities for Essential Service water
NEK's Essential Service Water system inspections encompassed four UT inspections (conventional and guided waves) and a VT of the excavated portions, which means 90% of total length of charging lines. Inspections were performed from 2003 to 2015.

That is equal to the maximum accessible length due to configuration of pipeline and the vicinity of the buildings that are penetrated by those pipelines. On the whole length of inspected charging pipelines, no degradation was detected. Visual inspections revealed a very good condition of protective bitumen coating on the pipelines, a 4-5 mm layer. Visual inspection of pipeline expansion joints also shows good condition. Ultrasonic examination has shown that there are no signs of pipeline damage.

Several smaller signs of corrosion were found on CC Heat Exchanger's SW discharge pipelines (the return water, back to the river). Potential propagation of those indications is going to be monitored by inspection of the same pipeline sections in the future.

Inspection activities for Diesel Generator Fuel Oil
In 2015, NEK performed visual and ultrasonic inspection of buried Diesel Generator fuel oil pipelines (DG1 and DG2). The inspection covered approximately 40% of the total length of pipelines. The inspection was conducted on two locations: within the concrete channel filled, by grain and in the vicinity of DG fuel tanks.

Visual inspection determined that the condition of bitumen coating is very good with a layer of 2-5 mm. Ultrasonic examination confirmed that there were no signs of pipeline damage. In addition, there were no signs of corrosion on the pipelines. The conclusion of the inspection was that pipelines were in very good condition and there was no need for the expansion of the inspection sample.

In the 2016 Outage, inspection of the underground Diesel Generator fuel oil tanks (DO100TNK001, DO100TNK002) and connecting pipelines was conducted. The main inspection method for outside surface corrosion inspection was Magnetic Flux Leakage – MFL. Additionally, thickness of the tank wall was precisely measured by ultrasonic scanning of the tank wall. Interior surfaces of both tanks were found to be without signs of corrosion damage. Both methods were demonstrated on a test sample prior to inspection of tanks and verified to be appropriate for detection.

Besides inspection of the tanks, connecting pipelines were planned to be inspected by long range ultrasonic guided waves method during the same outage. However, because of inaccessibility of the pipelines, ultrasonic probes could not be installed and the method of visual inspection of pipelines interior with the video borescope was performed instead. All inspection results showed that both tanks and connecting pipelines were in very good condition.

Inspection activities for Fire Protection
In 2015, NEK performed visual and ultrasonic inspection of concealed pipework on Fire Protection system. There were three inspection locations: two at the entrance to the concrete plate in DG1 and DG2 rooms and third location in the shaft JFO02. Visual inspection showed no degradation of anticorrosion protective coat. Ultrasonic guided wave method did not show any signs of corrosion damage on the pipelines.

4.1.3.2 Frequencies of inspection

NEK program TD-2Z, "*Buried and Underground Piping and Tanks*", Rev.3 [77] provides the following requirements concerning methods and frequencies for detection aging effects:

- Opportunistic inspections, such as visual testing whenever a buried or underground component become accessible for any reason (excavation);
- Directed inspections (visual supplemented with surface or volumetric - ultrasonic or magnetic flux leakage) as per NUREG-1801, Rev. 2, XI.M41 Tables 4 [18][18].

Opportunistic inspections in NEK are rare, since there was no need for excavation of the underground pipework or tanks. Only one occasion occurred in 2010, when portions of Essential Service Water System piping were excavated in order to install cathodic protection elements on the pipelines. NEK used this opportunity and performed inspection on accessible parts of the piping.

NEK performed a number of directed inspections, which means that the excavations were made for inspection purposes only. Those inspections were performed on the Essential Service Water, Diesel Oil and Fire Protection systems. System underground pipelines/tanks were partially excavated, cleaned and examined using conventional ultrasonic examination or long range guided ultrasonic examination and, especially for Diesel Oil tanks, magnetic flux leakage examination was carried out.

NEK performed the volumetric examination of the Essential Service Water system to the maximum extent possible before having entered the extended life time.

Diesel Oil tanks were inspected on all accessible surfaces from the inside using volumetric methods and Diesel Oil pipework was inspected for 40% of the total length of the buried pipework. As no significant degradation was found, NEK concluded that the condition of the Diesel Oil pipelines was good.

Buried portions of Fire Protection system were mostly replaced with High Density Polyethylene material. The remaining segments made of steel were examined at approximately 80 % of lengths, which is the maximum accessible lengths for the long range ultrasonic guided wave technology.

More details about inspections and inspection results are provided in the Section 4.2.

4.1.3.3 Acceptance criteria

The acceptance criteria for coated piping and tanks are that there should be no evidence of coating degradation or that the type and extent of coating degradation should be insignificant. If coated or uncoated metallic piping or tanks show evidence of corrosion the remaining wall thickness in the affected area shall be determined to ensure that the minimum wall thickness is maintained. Indications exceeding acceptance criteria (NUREG- 1801; Generic Aging Lessons Learned (GALL) Report, Rev. 2) must be reported in accordance with NEK Corrective Action Program [38].

The results of monitoring and inspections of concealed pipework and tanks at NEK showed no significant degradations. All inspected portions showed very good condition.

4.1.4 Preventive and remedial actions for concealed pipework

The industry practice is that buried piping and tanks are coated during installation with a protective coating system. The coating system include coal tar enamel with a fiberglass wrap and a craft paper outer wrap, a polyolefin tape coating or a fusion bonded epoxy coating to protect piping from being in direct contact with the aggressive soil environment. The cathodic protection system is used to mitigate corrosion, where pinholes in coating allow piping or components to be in contact with the aggressive soil environment. The cathodic protection imposes a current from an anode onto the pipe or tank to prevent the start of corrosion.

The effectiveness of coatings and cathodic protection system per standard industry practice is determined as monitoring the loss of material by conducting opportunistic bell hole examinations to visually examine the condition of external coating and wall thickness of piping and tanks. Any evidence on damaged wrapping or coating defects, such as coating perforation, peeling or other damage is an indication of possible corrosion damage to the external surface of piping and tanks which requires corrective action. Monitoring of pipe-to-soil potential and cathodic protection current are two possible additional parameters to determine the effectiveness of cathodic protection systems and thereby the effectiveness of corrosion mitigation measures [19].

Adverse indications found during monitoring of cathodic protection systems or during inspections are entered into the NEK's Corrective Action Program. Adverse indications may be leaks, material thickness less than minimum, the presence of coarse backfill with accompanying coating degradation within 6 inches of a coated pipe or tank and general or local degradation of coatings or the base material is exposed. Adverse indications that fail to meet the acceptance criteria will require the repair or replacement of the affected component. An analysis may be conducted to determine the potential extent of degradation observed. If adverse indications are detected, inspection sample sizes within the affected piping categories are doubled. If adverse indications are found in the expended sample, the inspection sample size should be again doubled. This doubling of the inspection sample size continues as long as necessary [19].

Based on results of inspections performed so far, no corrective actions were initiated, since no significant degradations or damages were found at NEK. Some smaller corrosion indications were found on the Component Cooling (CC) Heat Exchanger's SW discharge pipelines (return water back to the river), but no such that would require corrective actions to be performed. Future monitoring of those pipeline portions is planned to detect potential further degradations.

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

During the implementation of aging management program for concealed pipework, NEK did not find any relevant indications which would require actions to be taken. All inspected pipelines and tanks show very good condition. Several minor corrosion indications were found on the SW discharge pipelines which will be monitored within future inspections to make sure there is no propagation. Considering results from concealed pipework inspections, there was no need for changes to the aging management program for concealed pipework.

NEK concluded the first cycle of concealed pipework aging management inspections and is preparing for the second cycle focusing on the trending of the results from the first cycle, especially on those pipeline sections where indications of corrosion have been found.

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

a) Assessment of the ageing management processes

Krško NPP's general aging management program (MD-5) prescribes the TD-2Z program for concealed pipework. The TD-2Z "Buried and underground piping and tanks" program is in accordance with NUREG-1801 (GALL) Section XI.M41 and covers all the GALL and TPR requirements about concealed pipework. The purpose of this program is to implement aging management of buried and underground piping and tank surveillance. The program includes surveillance and preventive measures to mitigate corrosion by protecting the external surface of buried and underground steel piping and tanks. Surveillance and preventive measures are in accordance with standard industry practice based on National Association of Corrosion Engineers (NACE) Standards.

The most important interface program is TD-1Z1 "Open-cycle cooling water system program," which is also in accordance with NUREG-1801 (GALL) Section XI.M20. It covers aging management of internal surfaces of essential service water system components in the scope of concealed pipework. The purpose of this program is to increase safety and reliability of the Krško NPP by predicting, detecting, monitoring, testing, and mitigating open-cycle cooling water system components aging degradation.

b) Experience from inspection and assessment as part of its regulatory oversight

The regulator (SNSA) annually performs thematic inspections in the Krško NPP in order to check and assess the status of implementation of both programs that are relevant to ageing management of concealed pipework (TD-2Z and TD-1Z1). Experiences from inspections are the following:

- Essential Service Water (SW) System: more than 90% of all concealed pipework was inspected; no corrosion or mechanical damage was detected on inlet lines, but there was some minor degradation observed on SW outlet lines (reduction of wall thickness);

- Diesel Oil (DO) System: about 40% of the DO system pipeline was inspected; as no damage was observed and the bitumen anti-corrosive coat was in very good condition, no further inspections were carried out. Both buried tanks of the DO system and their connecting pipelines were inspected as well – inspections showed that the DO tanks and their pipelines were in excellent condition without any degradation;

- Fire Protection (FP) System: a part of the FP system with concealed steel pipes was inspected on 5 locations. Visual observation and ultrasound guided waves inspection showed no wall thickness reduction due to corrosion and no degradation of anti-corrosion protection.

c) The main strengths and weaknesses that have been identified either by the licensees or the regulator on the effectiveness of the SSC specific AMPs

Main strengths:

- The programs for ageing management of concealed pipelines are quite efficient, which is proved by a very good state of concealed pipework and buried tanks in the Krško NPP;
- Apart from the identification of degradation, the programs include further inspection of locations with identified damage indications in order to monitor and prevent potential propagation of these indications.

Main weaknesses:

- Technical implementing procedures for detection of ageing effects are not included in the TD-2Z "Buried and underground piping and tanks" program, although they should be developed and described in the program. Provisionally, that kind of procedures are provided by external contractors, which perform inspections. These contractor procedures are checked and approved by the Krško NPP. The Krško NPP is therefore obliged to include the implementing procedures in the next revision of the TD-2Z program.

d) The conclusions on the adequacy of the licensee's SSC specific ageing management programme(s)

The regulator's opinion about these two programs is that they are both generally well prepared and are adequately applicable for aging management on a respective matter. The weaknesses mentioned can be relatively easily eliminated and this will be done by the Krško NPP by the end of the year.

5 Reactor Pressure Vessel

5.1 Description of aging management programs for RPVs

Basic NEK programs for aging management of RPV are:

- TD-2E/4 "Inservice inspection program - the 4th inspection interval" [82]. This is a top-level program which includes the requirements of ASME XI [81] for the surveillance of reactor vessel base material welds, base material cladding and other auxiliary welds and structures forming part of pressure boundary.
- TD-2R "Reactor Head and BMI surveillance program" [87]. This program includes surveillance of reactor vessel closure including incore instrumentation and seal table weld.
- ED-5 "Reactor Vessel Irradiation Surveillance Program" [88]. This program includes material specimen based surveillance of reactor vessel beltline material embrittlement due to fast neutron fluence according to 10 CFR 50 Appendix H and ASTM E185 [89]. Additionally, this program covers the surveillance of neutron fluence by means of Ex-Vessel Neutron Dosimetry System after the completion of ASTM E185-82 based program.
- TD-2S "Inconel 600/82/182 Surveillance program" [90]. This program defines locations and required inspections for Inconel 600/82/182 material.
- Administrative and implementation procedures:
    - Implementation and administrative procedures for the execution of aging management programs are listed in chapters 6.0 and 7.0 of individual programs.
    - In addition to NEK procedures, qualified EPRI procedures and approved contractor procedures are used for UT inspections in accordance ASME XI [81], Appendix VIII.

5.1.1 Scope of aging management for RPVs

Reactor pressure vessel is schematically presented in Figure 5-1. The two major components of the reactor pressure vessel are the reactor vessel shell (including nozzles and lower head) and the reactor vessel closure head. The reactor vessel shell and head are attached together by means of a flange and 48 tensioned bolts. The reactor vessel shell is the original one from the time of plant construction in 1981, whereas the reactor vessel closure head was replaced preventively during the 2012 Outage to prevent potential Primary Water Stress Corrosion Cracking (PWSCC) issues related to Alloy 600/82/182 penetration welds on the original reactor vessel head. The material specification for the current reactor vessel head is ASME SA508, Grade 3, Class 1 (ASTM A 508). The head has 38 penetrations with Alloy 690 J-groove welds for Control Rod Drive Mechanisms (CRDM) (33 positions), Thermocouple Seal Assembly (3 positions), Head Vent Tube (1 position) and Inadequate Core Cooling Monitoring System (ICCMS) tube (1 position) (Ref. [91]).

The reactor vessel shell is presented in Figure 5-2. It consists of lower head, beltline region (lower shell, intermediate shell, upper shell), which also carries two inlets, two outlet nozzles and two safety injection nozzles, all aligned along the same centerline and vessel flange. The material specification of the lower head, lower shell, intermediate shell and upper shell is ASME SA 533, Grade B, Class 1. The material specification of the inlet nozzles, outlet nozzles, safety injection nozzles and vessel flange is ASME SA 508, Class 2. The weld disposition between various elements of the reactor vessel shell is depicted in Figure 5-2. The inner surface of the reactor vessel is coated with a stainless steel clad. The lower head has 36 penetrations with Alloy 600/82/182 welds for in-core instrumentation tubes.

a) Methods and criteria used for selecting components within the scope of aging management
Scoping and screening for NEK Systems, Structures and Components (SSC) including Reactor Pressure Vessel is performed in accordance with NRC 10 CFR 54 [12] and NEI 95-10 [16].

From this scope, according to the 10 CFR 54.21, only the passive SSCs that are not periodically replaced are subject to Aging Management Review, which includes Reactor Pressure Vessel.

b) Processes/procedures for the identification of aging mechanisms for different materials and components of the RPV.

In order to fulfil the requirements of 10 CFR 54, NUREG-1801 [18], Section IV.A2 is used for the identification of aging mechanisms for different materials and components of the RPV. However, as specified in Ref. [49], Section 05 »Reactor Pressure Vessels«, the aging mechanisms for the pressure boundary include the following reactor vessel components:

- The vessel lower, intermediate, upper shell and with pertaining welds;
- The vessel closure head and lower head including penetrations;
- Inlet, outlet nozzles and safety injection nozzles.

5.1.2 Aging assessment of RPVs

In this section, aging assessment of RPV is performed on the basis of aging mechanisms credited in NUREG-1801 [18] for the RPV components within the scope of this report as listed above.

Reactor Vessel Material Embrittlement Due To Irradiation

This aging mechanism is characterized by a decrease in material fracture toughness and by a shift in the transition temperature from ductile to brittle fracture, all due to fast neutron exposure (E>1 MeV) of the reactor vessel's ferritic steel. The measure of material fracture toughness is the upper-shelf energy obtained by a Charpy V-notch material specimen test. The measure of the transition temperature from brittle to ductile fracture is Reference Nil Ductility Temperature (RT_{NDT}), which is also obtained by a Charpy V-notch test. The applicable section in NUREG-1801 [18] covering this aging mechanism and the respective surveillance program is XI.M31. The scoping criterion as to which reactor vessels shall be included in the irradiation surveillance program is established in 10 CFR 50, Appendix H [65]. In its Section III.A it is specified that no material irradiation surveillance program is required for reactor pressure vessels for which the peak fast neutron fluence (E>1MeV) does not exceed 1.0×10^{17} n/cm^2 at the end-of-life (EOL). For vessels which do not meet these criteria, an irradiation surveillance program of the beltline materials according to ASTM E185 [89] must be established as required in Section III.B of Ref. [65]. Since NEK's RPV is above this figure, it has an ASTM E185 [89] based irradiation surveillance program of the reactor vessel beltline materials. Beltline materials are defined in 10 CFR 50, Appendix G [66], Section II.F, and for NEK this includes lower, intermediate and upper shell with related welds and heat affected zones. For testing purposes, representative material specimens of base material, weld material and heat affected zone are available. The material of reactor vessel inlet nozzles, outlet nozzles and safety injection nozzles SA508 has a significantly lower initial RT_{NDT} than the beltline material. An increase of the RT_{NDT} during the lifetime is also expected to be smaller compared to the beltline material due to a relatively small neutron fluence at nozzle locations (< $1.0\text{x}10^{17}$ n/cm^2). For these reasons aging due to the irradiation material embrittlement is not critical for the nozzle material SA508.

ASTM E185 [89] defines a surveillance program based on the material specimen testing. Such a program consists of periodically withdrawing capsules attached to the inner surface of RPV, which contains material specimens of the vessel beltline material (base material, weld and heat affected zone) and dosimeters. The most relevant test for determining the extent of the embrittlement aging mechanism is a Charpy V-notch test of irradiated steel specimens. One important result of the Charpy V-notch test is to measure increase in transition temperature from ductile to brittle fracture. This is used to determine the Adjusted Reference Nil Ductility Temperature (ART_{NDT}) at 60-year EOL according to RG-1.99 [41]. The other important result of the Charpy V-notch test is to measure drop in the upper-shelf impact energy as a parameter of material toughness.

The acceptance criterion for the vessel beltline material is sufficient upper-shelf impact energy at the EOL as defined in 10 CFR 50, Appendix G [66], Section IV.A.1. Also the results of the Charpy V-notch testing combined with the calculated peak neutron fluence at the vessel shell wall at 60-year EOL are used to determine the limiting pressure-temperature operating curves for the reactor vessel per 10 CFR 50, Appendix G [66], Section IV.A.2. For protection against Pressurized Thermal Shock (PTS), the acceptance criterion is given in 10 CFR 50.61 [52], whereby the reference PTS temperature (RT_{PTS}) of the beltline material, which is equal to ART_{NDT} at EOL must be less than the screening criterion of [52], section (b).(2).

Additionally, during the 2010 Outage, NEK installed an Ex-Vessel Neutron Dosimetry system. This system is used to control the analytically-established neutron fluence, which was used in determining the pressure-temperature operating limits at 60-year EOL, after the final capsule withdrawal in the 2012 Outage.

Mechanical Fatigue of Reactor Vessel

Mechanical fatigue is related to repetitive operating transients with oscillating temperature/pressure conditions. Such transients cause alternating stresses in the material and if repeated cyclically, may result in the initiation of cracks. Mechanical fatigue of components is affected by the primary coolant environment in a way that could significantly reduce the fatigue resistance of the pressure boundary components. A special subcase of mechanical fatigue is the fatigue crack growth, where an existing flaw combined with cyclic oscillating pressure/temperature loading may grow and potentially affect the pressure boundary. The applicable section in NUREG-1801 [18] covering this aging mechanism and the respective monitoring program is X.M1.

For Class 1 structures, mechanical fatigue as a consequence of cyclic transients is incorporated in the design of the plant by ASME III [92]. Therefore, the elements analyzed for this aging mechanism include lower shell, intermediate shell, upper shell, vessel flange, lower head, vessel closure head, inlet, outlet, safety injection nozzles and associated welds. Fatigue analysis of the reactor vessel pressure boundary, as a Class 1 structure, is governed by ASME III [92], NB-3200 and takes into account cyclic stress loadings due to Normal Operating Condition transients and Upset Condition transients, as defined in NEK Design Transient Specification [93] and NEK Technical Specifications [48]. The acceptance criteria for mechanical fatigue is defined in NB-3222.4, and this is the Cumulative Usage Factor (CUF), which must be less than one (CUF<1). CUF is determined during the plant design process following the provisions of NB-3222.4. The number of Normal/Upset Condition transients listed in NEK Technical Specifications [48] is used as an input for the determination of CUF. Therefore, in order to assure CUF<1, the actual number of individual Normal/Upset/Test Condition transients experienced during plant lifetime shall be less than the number defined in the NEK Technical Specifications [48]. This is the acceptance criteria used in NEK to limit mechanical fatigue in the reactor vessel.

However, the CUF calculated with the design number of transients from TS [48] does not include the environmental effects on component fatigue. For this purpose an environmentally adjusted CUF_{en} is calculated for the critical locations using an additional environmental factor F_{en} per NUREG/CR-6717 [94]. The critical locations are determined in NUREG/CR-6260 [95] and include inlet, outlet and safety injection nozzles. The acceptance criterion is then $CUF_{en}< 1$. Results for environmentally adjusted CUF_{en} are presented in Ref. [96], where it is demonstrated that for the critical locations in reactor vessel CUF_{en} is <1 and not substantially larger than CUF at 60-year EOL. It is concluded that the limiting number of occurrences of Normal and Upset Condition transients listed in TS ([48], Table 4.7-1) is a credible acceptance criterion for mechanical fatigue including environmental effects, which ensures both CUF<1 and CUF_{en}<1.

Fatigue crack growth as a subcase of mechanical fatigue is an aging mechanism that is evaluated according to ASME XI [81] when flaws of sufficient size are present in the material. Flaws discovered during in-service inspection and exceeding the limiting flaw size per ASME XI [81], Section IWB-3500 are evaluated for acceptability using the fracture mechanics criteria from ASME XI [81], Section IWB-3600 and using the fatigue crack growth mechanism model from ASME XI [81], Appendix A, taking into account Normal, Upset, Emergency and Faulted Condition transients. Alternatively, in order to avoid performing fracture mechanics analysis for every flaw that may be revealed during in-service inspection and exceeds the limits of IWB-3500, a plant specific Flaw Evaluation Handbook [97] was prepared. This Handbook contains flaw evaluation charts presenting limiting flaw depths at various time spans. The acceptance criterion is the found-flaw depth exceeding the limiting flaw size per ASME XI [81], Section IWB-3500. It should be less than the limiting-flaw depth in the flaw evaluation chart for a given time span.

Primary Water Stress Corrosion Cracking (PWSCC) in Components Containing Alloy 600/82/182

Primary water SCC is an intergranular cracking mechanism occurring in structures exposed to high stresses due to RCS pressure, primary water environment combined with high temperatures and having susceptible microstructures. In a PWR reactor vessel this degradation mechanism is important for welds containing Alloy 600/82/182, which connect ferritic steel components to stainless steel components. Such welds are also referred to as Dissimilar Metal Welds (DMW). In NEK, these are RPV inlet/outlet nozzle-to-safe-end butt weld (4 welds), safety injection nozzle-to-safe end (2 welds) and BMI (Bottom Mounted Instrumentation) penetration welds on the lower head. The applicable section in NUREG-1801 [18] covering this aging mechanism and the respective monitoring program is XI.M11B.

PWSCC in the components containing Alloy 600/82/182 is controlled by surveillance of potential flaws through in-service inspection. Since the limiting flaw size criteria of ASME XI [81], Section IWB-3500 does not apply to potentially-discovered surface flaws in the material containing Alloy 600/82/182, such flaws are immediately evaluated for acceptability using fracture mechanics criteria from ASME XI [81], Section IWB-3600 and using fatigue crack growth mechanism from EPRI MRP-115 [98] taking into account Normal, Upset, Emergency and Faulted Condition transients. Alternatively, in order to avoid performing fracture analysis for every flaw that may be revealed during in-service inspection, a plant specific Flaw Evaluation Handbook [99] was prepared. This handbook contains flaw evaluation charts presenting limiting flaw depths at various time spans. The acceptance criterion is that the found-flaw depth is less than the limiting-flaw depth in the flaw evaluation chart for a given time span.

Extensive operating experience has been gathered industry-wide throughout the years of NPPs' operation. The pivotal examples of PWSCC that prompted nuclear industry to issue inspection guidelines, as the safety consequences of inadequate inspections could have been significant, were the V.C. Summer through-wall axial cracking event in 2000 and the finding of three and four flaws in nozzle-to-safe-end welds at Ringhals units 3 and 4, respectively, between 1999 and 2001. Other cases of indications found due to PWSCC in the reactor pressure vessel DMW include Davis Besse plant 2008, Salem plant 2008, and Seabrook plant 2009.

Related to the operating experience, the EPRI Materials Reliability Program (MRP) issued MRP-139 [100] in 2005 provides industry guidance for volumetric and visual inspections of unmitigated and mitigated butt welds in PWR primary coolant systems. In 2005, the ASME (American Society of Mechanical Engineers) also approved the development of a Code Case on appropriate inspection requirements to address PWSCC in Class 1 butt welds containing Alloy 82/182. This case was later numbered as Code Case N-770 and was revised in 2009 to address NRC concerns, upon which Code Case N-770-1 [101] was issued later that year. Code Case N-770-1 contains requirements for inspection of unmitigated as well as Alloy 82/182 reactor coolant system (RCS) butt welds mitigated by certain techniques. The NRC incorporated ASME Code Case N-770-1 by reference into 10 CFR 50.55a [67], section (g)(6)(ii)(F) in June 2011 and in August 2017 Code Case N-770-2 [102]. Subsequently, further revisions of Code Case N-770 were issued but none of these is incorporated in 10 CFR 50.55a. ASME issued other Code Cases related to the examination of welds and components fabricated from Alloy 600/82/182 and susceptible to PWSCC, among which Code Case N-722-1 [103] and N-729-4 [104] were incorporated directly into 10 CFR 50.55a.

5.1.3 Monitoring, testing, sampling and inspection activities for RPVs

Reactor Vessel Material Embrittlement Due to Irradiation

The respective NEK program is ED-5 [88]. The initial program consists of six capsules containing material testing specimens (Charpy V-notch and tensile) of the RPV beltline base material, weld material, heat affected zone material and dosimetry. Based on ASTM E185 [89] requirements, four out of six capsules were withdrawn according to a schedule laid out in NEK Technical Specifications [48], Table 3.4-6. The last capsule was withdrawn during the 2012 Outage. This concludes the capsule withdrawal program for the 60-year EOL plant operation. Of the two remaining capsules, one is left in the Reactor Vessel and the other was moved to the Spent Fuel Pool. These two capsules may be used later on to obtain experimental data for a hypothetical plant lifetime extension beyond 60-years EOL.

After the conclusion of the capsule-based program, Ex-Vessel Neutron Dosimetry (EVND) system is used to validate the analytically established neutron fluence, which was used in determining the pressure-temperature operating limits at 60-year EOL. Ex-Vessel Neutron Dosimetry system is placed between the outer wall of the reactor vessel and the surrounding biological concrete shield and consists of five dosimeters placed at three azimuthal locations around the perimeter of the reactor vessel outer wall. Given that NEK core is 1/8 symmetrical, the dosimeters are placed at azimuthal locations that correspond to 0°, 15° and 30° in a 1/8 quadrant. The withdrawal schedule for the Ex-Vessel Neutron Dosimetry is defined in NEK Technical Specifications [48], Table 3.4-7, and is based on the recommendations of the EVND supplier.

Mechanical Fatigue of Reactor Vessel

Mechanical fatigue of the reactor vessel is monitored by counting Normal and Upset Condition transients. This is documented in a technical report per plant procedure ADP-1.2.010 [105] after each fuel cycle. The acceptance criterion is the cumulative number of each Normal and Upset Condition transient and should be less than the allowed number per NEK Technical Specifications [48], Table 4.7-1. The severity of monitored transients is also verified to be bounded by NEK Design Transient Specification [93].

The program to discover potential flaws in the reactor vessel material is TD-2E/4 [82], »In-service Inspection Program« (ISI). The ISI program for fourth inspection interval as required by NEK Technical Specification SR 3.0.5 is based on the requirements of the 2007 Edition with the 2008 Addenda of ASME XI [81], referenced in paragraph (b) of the 10 CFR Part 50.55a [67] in June 2011. The duration of an in-service inspection interval is 10 years.

The ISI Program plan lists the components, the NDE methods and the schedule of examinations necessary for implementation of the program during the corresponding ten-year interval. In NEK, ISI program plan includes the NDE examinations of Reactor Vessel Welds and pressure retaining bolting.
ISI program includes the following requirements for non-destructive examinations of Reactor Vessel as prescribed in ASME XI [81]:

- Visual examinations (VT-1, VT-2 and VT-3) should be performed in accordance with the requirements of IWA-2210 of Section XI and Article 9 of ASME V [106].
- Surface examination indicates the presence of surface discontinuities. It may be conducted using magnetic particle (MT), liquid penetrant (PT) or eddy current (ET) method.
    - Magnetic particle (MT) examination should be conducted in accordance with ASME V [106], Article 7.
    - For nonflourescent particles the visible light intensity required is 50 fc. Alternatively, light shall be sufficient if the examination can resolve standard test chart characters as described for VT-1 in ASME XI, IWA-2210 [81].
    - Liquid penetrant (PT) examinations should be conducted in accordance with ASME V [106], Article 6.
    - For visible dye penetrant, the visible light intensity required is 50 fc. Alternatively, light should be sufficient if the examination can resolve standard test chart characters as described for VT-1 in ASME XI, IWA-2210 [81].
    - Eddy Current (ET) examination for detection of surface flaws should be conducted in accordance with ASME XI, Appendix IV [81].
- Volumetric examination indicates the presence of discontinuities throughout the volume of material and may be conducted from either inside or outside surface of component:
    - Ultrasonic examination (UT) should be conducted in accordance with ASME XI, Appendix I [81], which provides standards for UT implementation related to inspection items. Certain reactor vessel welds and bolts requiring UT should be examined per mandatory ASME XI, Appendix VIII [81]. NE Krško should use Appendix VIII as implemented by the industry Performed Demonstration Initiative; EPRI PDI Program (Appendix VIII as implemented by the industry Performance Demonstration Initiative - EPRI proprietary) for ultrasonic examination systems.

NEK follows the requirements for examinations of Reactor Vessel specified in ASME Section XI [81], Tables IWB-2500-1. The provisions for the following Examination Categories and Items are defined: Exam. Cat. B-A, Item B1.11 Circumferential Shell Welds; Exam. Cat. B-A, Item B1.12 Longitudinal Shell Welds; Exam. Cat. B-A, Item B1.30 Shell-to Flange Weld; Exam. Cat. B-D, Item B3.90 Nozzle-to Vessel Welds; Exam. Cat. B-D, Item B3.100 Nozzle Inside Radius Section; Exam. Cat. B-G-1, Item B6.10 Closure Head Nuts; Exam. Cat. B-G-1, Item B6.20 Closure Head Studs; Exam. Cat. B-G-1, Item B6.40 Treads in Flange; Exam. Cat. B-G-1, Item B6.50 Closure Washes; Exam. Cat. B-K, Item B10.10 Welded Attachments; Exam. Cat. B-N-2, Item B13.60 Interior Attachments Beyond Belting; Exam. Cat. B-O, Item B14.21 Welds in control Rod Drive Housing; Exam. Cat. B-O, Item B14.21 Welds in In-Core Instrumentation Nozzle Housing; Exam. Cat. B-P, Item B15.10 Pressure Retaining Boundary.
ISI program [82] uses the following generically approved ASME Code Cases related to Reactor Vessel specified in Regulatory Guide 1.147 [107]:

- ASME Code Case N-460 [108],
- ASME Code Case N-613-1 [109],
- ASME Code Case N-648-1 [110].

Flaws that are revealed during in-service inspections and exceed the limits of ASME XI [81], IWB-3500, are evaluated per Flaw Evaluation Handbook [97]. This handbook contains a series of flaw-depth-vs.-flaw-shape charts for individual components of the reactor vessel. Each chart contains limiting curves for different time spans (10 years, 20 years, 30 years) which effectively present limiting depth of a flaw which if left growing for 10, 20 or 30 years would reach a depth corresponding the fracture mechanics limiting criteria of ASME XI [81], IWB-3600. Consequently, flaws with depth below this limiting curve are acceptable for continued operation without immediate repair for the respective time span.

The frequency of reactor vessel material inspection is defined in accordance with ASME Section XI, Table IWB-2500-1 and NEK In-service Inspection program TD-2E/4. The pre-service inspection (PSI) was done in 1979, second inspection took place in 1992, followed by successive 10-year inspection intervals. Last full scope inspection of RPV, which included ultrasonic and visual inspection was performed in 2010 with the following conclusions:

- Ultrasonic examinations of the shallow volume of RPV shell welds revealed 4 flaw-like indications in all inspected shell welds. The flaw-like indications are found to be metallurgical inclusions of non-volumetric and volumetric type (depending on the signal shape). Indications were assessed to ASME Code acceptance standards and found to be allowable;
- Ultrasonic examination of shell welds volume from 1/3 of inspection volume to the whole component thickness revealed 4 flaw-like indications which are assessed to ASME Code acceptance standards and found to be allowable. In addition, signals at OD surface are found at welds BW-1, BW-2, BW-4 and RPV supports and are determined to be generated by OD surface geometry;
- Based on the comparison of RPV-ISI 2001 and 2004 results versus RPV-ISI 2010 results, it can be concluded that three new indications are found to be reportable and are directly related to the application of the new more sensitive tip-diffraction technique with reporting and evaluation methodology included in the procedure;
- Furthermore, the indication comparison shows that there is no flaw growth for indications previously reported;
- Visual examination of Reactor Pressure Vessel interior showed few reportable indications and one non-reportable indication on examined locations, on core support lugs (scratches and wear of surface) and vessel bottom (presence of foreign material). Visual examination of other examined area of Reactor Pressure Vessel showed no indications.

<u>Primary Water Stress Corrosion Cracking (PWSCC) in Components Containing Alloy 600/82/182</u>
The respective NEK programs are TD-2S [90], TD-2R [87], and TD-2E/4 [82]. TD-2S is an overall program for Alloy 600/82/182 surveillance. TD-2S lists locations where Alloy 600/82/182 is found and defines additional scope and frequency of inspection specifically for Dissimilar Metal Welds (DMW) through ASME Code Cases N-722-1 [103], N-729-4 [104] and N-770-2 [102]. TD-2E/4 is a general inservice inspection

program which also includes inspections, techniques, scope and frequency for DMW on reactor vessel nozzles. TD-2R includes inspections, techniques, scope and frequency for welds containing Alloy 600/82/182 on Bottom Mounted Instrumentation penetrations.

NEK employs mandatory augmented programs for DMWs on Reactor Vessel Nozzle-to-Safe End Welds and Bottom Mounted Instrumentation penetrations based on provisions from ASME Code Cases N-770-2 and N-722-1, as required by 10 CFR 50.55a(g)(6)(ii)(F) and 10 CFR 50.55a(g)(6)(ii)(E) [67], respectively. NEK's two Hot leg DMWs are the Inspection Item A-2; Unmitigated Butt weld at Hot Leg operating temperature ≤ 329°C. Two Cold leg DMWs are the Inspection Item B; Unmitigated butt weld at Cold Leg operating temperature ≥ 274°C and < 304°C. Based on ASME Code Case N-770-1, volumetric examination (ultrasonic examination UT) in accordance with ASME XI, Appendix VIII [81], should be performed for the following Inspection Items: Every 5 years for the Inspection Item A-2, every second inspection period not exceeding 7 years for the Inspection Item B. For those DMWs, visual examination is also prescribed based on Code Cases N-770-2 and N-722-1. Visual examination (VE) should be performed on outer surfaces of DMWs for evidence of pressure boundary leakage and corrosion on adjacent ferritic steel components. Frequency for examination based on N-770-2 is each refueling outage for the Inspection Item A-2 and once per interval for the Inspection Item B. An ultrasonic examination (UT) performed from the component's interior, in accordance with requirements prescribed by the ASME Code Case N-770-2, Table 1 and ASME XI, Appendix VIII and should be acceptable in lieu of visual examination (VE) required by the ASME Code Case N-770-2.

The frequency of inspection for the DMWs on Bottom Mounted Instrumentation penetrations is prescribed by ASME Code Case N-722-1, and the examination should be performed every second refueling outage. Acceptance standards for DMWs based on examination method are defined in ASME Code Case N-770-2: for VE, paragraph 3140, and for UT paragraph 3130 should be applied. Acceptance criteria for Class1 PWR components containing Alloy 600/82/182 are prescribed by ASME XI, IWB-3522 as requested by ASME N-722-1. Flaws in DMW welds that require analytical justification are evaluated per Flaw Evaluation Handbook [99] for nozzle-to-safe end DMWs. This handbook contains a series of flaw-depth *vs.* flaw-shape charts for nozzle-to-safe end welds, which contain limiting curves for different time spans (6 months, 12 months, 18 months, 36 months). Flaws with depth below this limiting curve are acceptable for continued operation without immediate repair for a respective time span.

During the 2015 Outage, volumetric and surface examinations of RPV Nozzle-to Safe End weld inner surface were performed following the frequency of examinations defined by ASME Code Case N-770-1, which was mandatory at the time. In order to prove that no surface breaking defects are present that could lead to PWSCC, eddy current method was used for surface examination. Based on the RPV ultrasonic examination during the 2015 Outage, the following conclusions are drawn:

- Ultrasonic examinations of nozzle dissimilar metal welds confirmed five (5) historic indications. These were assessed according to ASME code acceptance criteria and found to be allowable. Considering the indications locations and their ultrasonic signal responses, the indications can be interpreted as metallurgical indications within the dissimilar weld material and related heat affected zone as well as at the cladding interface;
- Eddy current examinations of nozzle dissimilar metal welds and safe end-to-pipe welds revealed no reportable indications;
- By comparing the RPV-ISI 2004 and 2010 results versus present examination results, it can be concluded that no new indications are found within the inspected nozzle welds and adjacent base metal. Furthermore, the indication comparison shows that there is no flaw growth for indications previously reported;

During the 2016 Outage visual examination (VE) of RPV Nozzle-to Safe End welds and Bottom Mounted Penetrations in accordance with ASME Code Case N-770-1 and N-722-1 was performed. Visual examination (VE) on other surfaces of DMWs made from Alloy 600/82/182 showed no evidence of pressure boundary leakage and corrosion on adjacent ferritic steel components.

5.1.4 Preventive and remedial actions for RPVs

Reactor Vessel Material Embrittlement Due to Irradiation

The results of surveillance capsule testing, which are used to determine ART_{NDT} are incorporated into the plant operating limitations, i.e. pressure-temperature limiting curves, which are included in NEK Technical Specifications [48]. If an out-of-limit condition is incurred during plant operation, ASME XI, IWB-3730 specifies that an engineering evaluation of the beltline region structural integrity is performed according to ASME XI [81], Appendix E. Regarding beltline material upper-shelf impact energy, in case the acceptance criterion of 10 CFR 50 Appendix G [66] is not met, ASME XI, IWB-3730 specifies that ASME XI, Appendix K contains procedures that may be used to demonstrate protection against failure of embrittled material.

Mechanical Fatigue of Reactor Vessel

In case the CUF exceeds Construction Code [92] limit, the following actions are proposed per GALL report, [18] Section X.M1:

- Repair/replacement of the affected component;
- A more rigorous analysis of the affected component to demonstrate that the design limit is not exceeded during the lifetime of the plant.

Alternatively, if CUF exceeds the Construction Code [92] limit, ASME XI [81], IWB-3740 determines that fatigue assessment per Appendix L of [81] may be performed, which essentially allows the use of later Editions and Addenda of ASME III [92] instead of the plant original edition of Construction Code.

Primary Water Stress Corrosion Cracking (PWSCC) in Components Containing Alloy 600/82/182

The following preventive and remedial actions of PWSCC are established:

- Replacement of components with materials more resistant to PWSCC: Reactor vessel closure head was replaced in NEK, regardless of the fact that there were no flaw indications in Alloy 600/82/182 welds of the closure head. The new head does not contain Alloy 600/82/182 components;
- Weld Overlay: The weld overlay process consists of applying an annulus of Alloy 52-type weld material on the outside of a circular geometry component over the susceptible Alloy 82/182 DMW;
- Weld Inlay/Onlay: The inlay process consists of excavating a small portion of the susceptible material on the inside diameter surface of the circular geometry component weld and applying Alloy 52/152 material in its place to form a more corrosion-resistant barrier between the Alloy 82/182 weld and butter materials and reactor coolant. With the onlay process no excavation of material is performed prior to applying the barrier of Alloy 52/152 material. During the implementation of these methods only a small layer of 2 to 3 weld beads would be deposited on the inside surface of the circular geometry component;
- Mechanical Stress Improvement Process (MSIP): The MSIP process modifies the existing inside diameter residual tensile stresses in the weld metal and heat-affected zone of butt welds in circular geometry components. The load is applied to the outside surface of the circular geometry components with a large two-piece mechanical hydraulic clamping device connected by two pairs of tangentially positioned studs. Loading plastically strains the pipe causing the pipe diameter to decrease in the region under the clamps. In the nearby weld and counterbore region, the plastic strain caused by the clamps generates compressive residual stress around the weld inside dimeter and counterbore region.

5.2 Licensee's experience of the application of AMPs for RPVs

Reactor Vessel Material Embrittlement Due to Irradiation

The existing reactor vessel material surveillance program for this aging mechanism provides sufficient material and dosimetry data to determine the pressure-temperature operating limits for the reactor vessel at 60-year EOL. The results of Charpy V-notch test from the last withdrawn specimen capsule were used to determine NEK Technical Specification pressure temperature operating limits at 60-year EOL. Also based on the results from the last withdrawn specimen capsule, the upper-shelf energy at 60-year EOL was found to satisfy the acceptance criterion of 10 CFR 50, Appendix G [66] and the RT_{PTS} was found to satisfy the acceptance criterion of 10 CFR 50.61 [52] for Pressurized Thermal Shock.

Mechanical Fatigue of Reactor Vessel

The last available transient count relevant for the evaluation of mechanical fatigue of Class 1 components (which also includes reactor pressure vessel) is documented in [47], Table 4-1A, B (Normal, Upset) and covers the data until the end of cycle 28 (Autumn 2016). the current number of occurrences of all Normal and Upset Condition transients is below the TS limit ([48], Table 4.7-1). Bilinear extrapolation of the current number of occurrences of Normal, Upset and Test Condition transients to 60-year EOL is also performed and the extrapolated values are also below the TS limit ([48], Table 4.7-1).

Primary Water Stress Corrosion Cracking (PWSCC) in Components Containing Alloy 600/82/182

Implementation of ISI programs TD-2E/4 [82], TD-2S [90], TD-2R [87] for this aging mechanism provides sufficient control over PWSCC aging mechanism. Until now NEK has no experience with indications related to PWSCC. The programs for managing this mechanism have been augmented with revised 10 CFR 50.55a requirements originating from industry experience regarding PWSCC. During the 2012 Outage, NEK completed the project of Reactor Vessel Head (RVH) replacement. The new RV head contains no Alloy 600/82/182 components. All components of the new RV head are non-susceptible to PWSCC.

5.3 Regulator's assessment and conclusions on ageing management of RPVs

a) Assessment of the ageing management processes

Ageing management for reactor pressure vessel is covered with the main program TD-2E / 4 "In-service inspection program - the 4th inspection interval", which is prepared in accordance with ASME section XI standard. Additionally, there are the programs for control of particular elements of the reactor pressure vessel such as the reactor pressure vessel head and BMI penetrations surveillance program TD-2R "Program nadzora glave reaktorske posode in BMI penetracij", the reactor pressure vessel irradiation surveillance program ED-5 and Inconel 600/82/182 Surveillance Program TD-2S.

Systematic identification and evaluation of internal and external ageing management experiences are in place. In the past, the Krško NPP addressed different experiences, such as primary water stress corrosion cracking and based on that some important modifications i.e. the reactor pressure vessel head replacement and structural weld overlays on pressurizer were implemented.

The SNSA estimates that the ageing management programs are well developed and appropriately implemented. Adequate control of the materials and monitoring of the embrittlement of the reactor pressure vessel have been established. Generally speaking, the reactor pressure vessel is in good condition; no significant deviations have been identified.

b) Experience from inspection and assessment as part of its regulatory oversight

The ageing management program implementation of reactor pressure vessel is controlled by the SNSA during regular and thematic inspections.

In-depth inspection was performed in 2017 to assess the aging management of the reactor pressure vessel. The assessment of aging management programs, implementation of controls, testing, sampling and inspections of the reactor pressure vessel were the main topics of this inspection. The implementation of preventive and corrective measures and experiences of the plant with specific programs for monitoring the aging of the reactor pressure vessel were also discussed.

It was concluded that the Krško NPP has developed and implemented appropriate programs and procedures with methods for identification and monitoring the effects of aging. The requirements for preventive and corrective measures are also established. The NPP Krško did not identify unacceptable degradations that would affect the safe operation of the plant. All perceived deviations are recorded in the framework of the implementation of the corrective program. Based on the analyses and specifications of the applied standards and procedures, appropriate preventive and corrective actions were carried out.

The Krško NPP, as part of the monitoring of operational experiences in nuclear industry and according to the WENRA recommendations, analyzed the possibility of hydrogen flakes in the base material of reactor pressure vessel. Based on available information on the technological process of producing material for the reactor pressure vessel (Combustion Engineering manufacturer), the heat treatment and the results of the performed pre-service inspection, the Krško NPP concludes that the problem of the appearance of hydrogen flakes in the basic material of the reactor pressure vessel is not expected. Additionally, in 2013 the Krško NPP carried out a preventive inspection of the reactor pressure vessel calibration blocks using the UT method. No indications were found.

For all reportable indications on Reactor Vessel pressure boundary components found during the implementation of In-Service inspections (see section 5.1.3), no process of indication propagation was observed.

c) The main strengths and weaknesses that have been identified either by the licensees or the regulator on the effectiveness of the SSC specific AMPs

Main strengths:

- Programs for ageing management of the reactor pressure vessel are well developed and appropriate implemented;
- Major modifications are preventively implemented to exclude the issue of primary water stress corrosion cracking;
- The adequate control of the reactor pressure vessel materials and monitoring of the embrittlement of the reactor pressure vessel have been established.

d) The conclusions on the adequacy of the licensee's SSC specific ageing management programme(s)

The reactor pressure vessel is the most important SSC in terms of ageing. The Krško NPP has prepared and properly implemented aging management programs for monitoring, testing and inspection of the ageing processes of the reactor pressure vessel. The replacement of the reactor pressure vessel head in 2013 was done in the scope of preventive maintenance to exclude the issue of primary water stress corrosion cracking. Depending on the ageing management experiences, it was verified that the materials of the reactor pressure vessel are adequate regarding hydrogen flakes. Appropriate ageing management program implementation and preventive actions including modifications connected with the reactor pressure vessel ensure, that the integrity of the reactor pressure vessel and reactor core is not compromised.

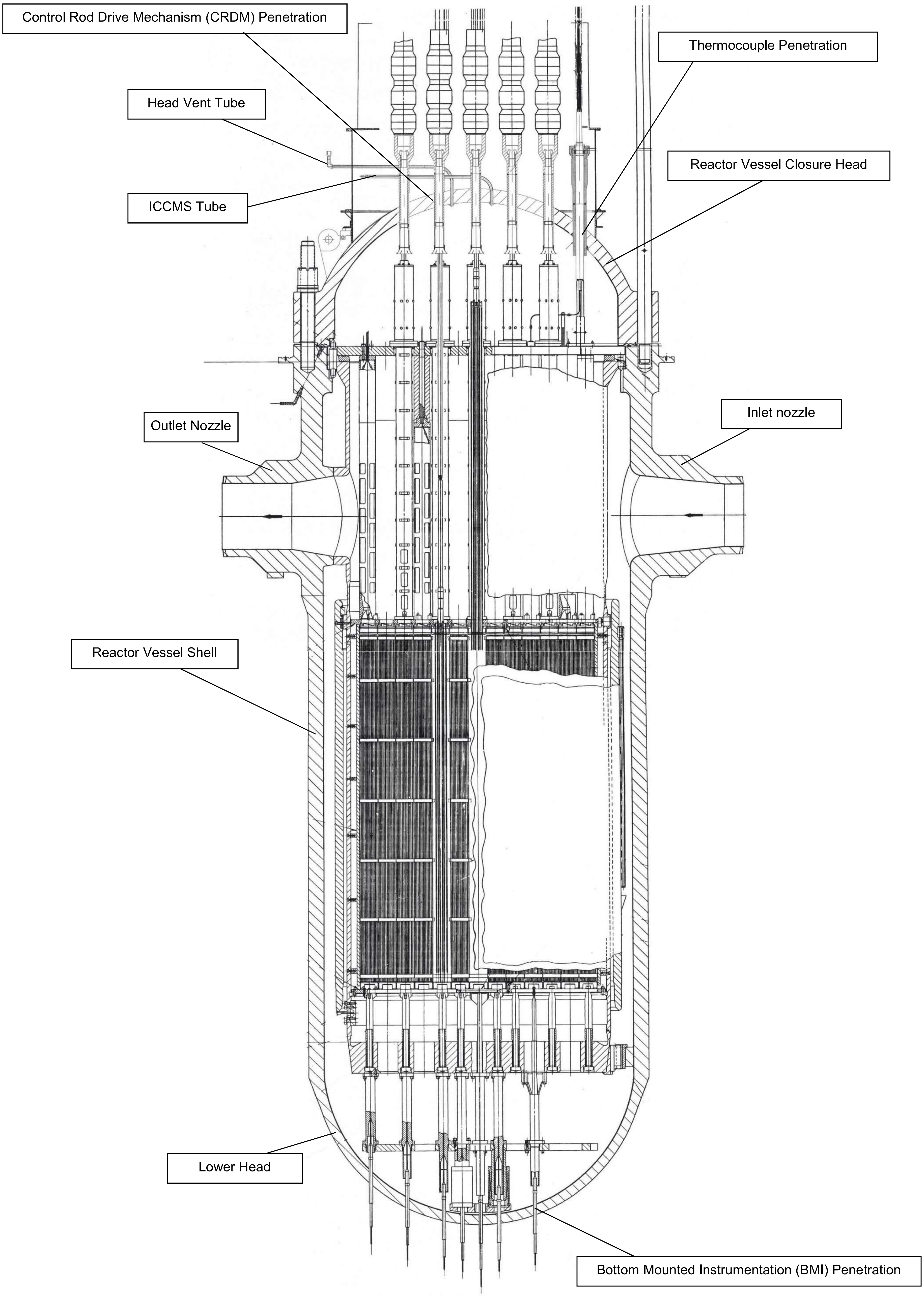

Figure 5-1: Reactor Pressure Vessel Schematic

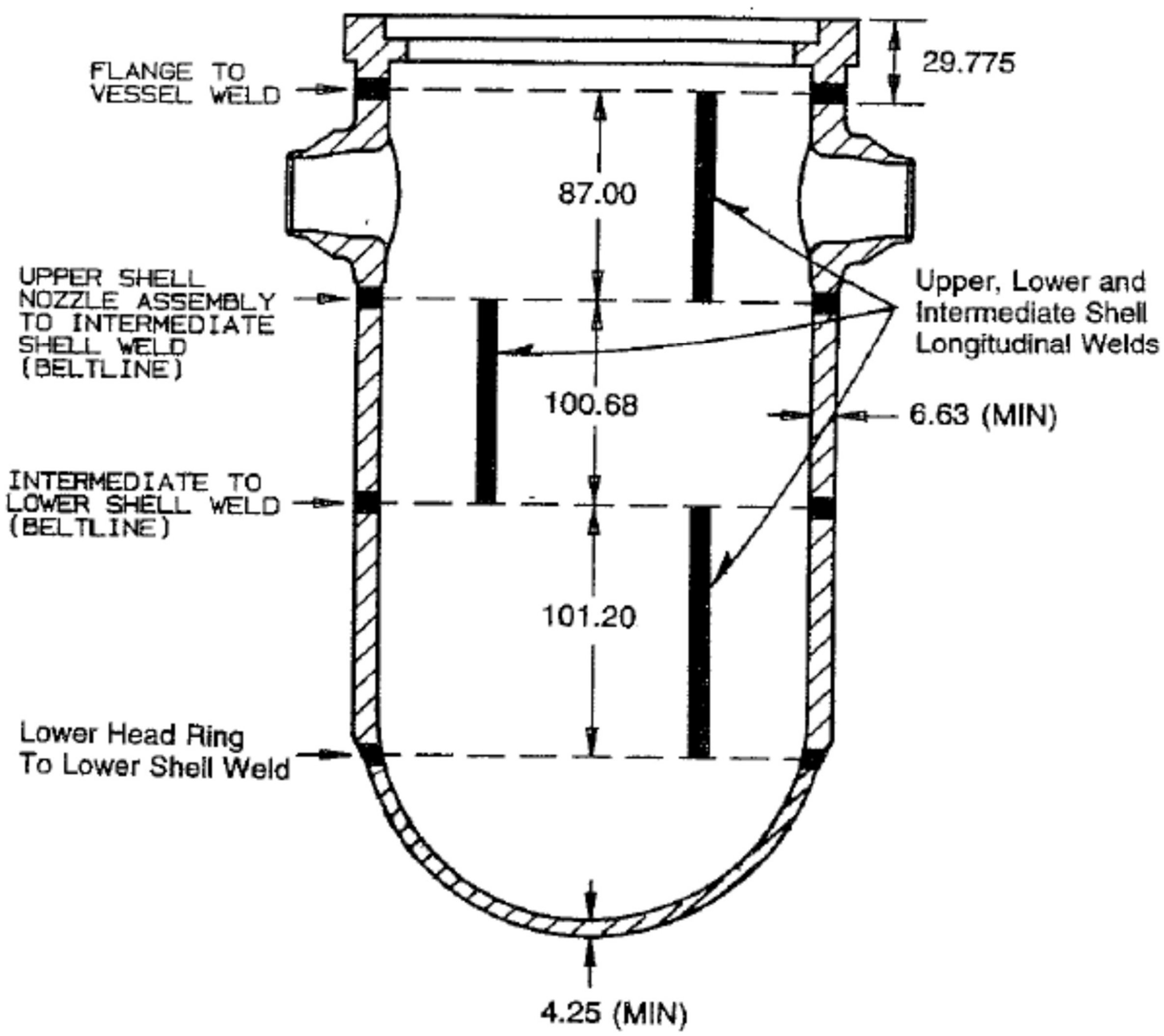

Figure 5-2: Reactor Vessel Shell

6 Calandria/pressure tubes (CANDU)

Not applicable for the Krško NPP.

7 Concrete containment structures

7.1 Description of aging management programs for concrete structures

7.1.1 Scope of aging management for concrete structures

No concrete containment structure is provided for NEK, but this chapter is used for a review of the aging management program for the Shield building.

The Shield building is a concrete structure that surrounds a (self-standing) steel containment, and it is designed as a relatively leak tight structure. This permits the use of the containment annulus ventilation and filtering system for minimizing escape of radioactive particles to the environment by maintaining a slightly negative annulus air pressure. At the same time it provides the principal biological shield, protection for the steel containment vessel from weather and external impacts. The concrete Shield building is shown in Figure 7-1.

The Shield Building is NRC RG 1.29, Seismic Category I structure. It consists of the reinforced concrete cylindrical wall, spherical segment dome and foundation mat.

The concrete Shield building is in the scope of aging management due to its important safety functions and provides:

- The principal biological shielding;
- Protection of the containment vessel from weather and external impacts;
- A relatively leak tight structure to permit the use of the containment annulus ventilation and filtering system for minimizing the escape of radioactive particles into the environment by maintaining a slightly negative annulus air pressure.

The NEK Steel Containment is ASME B&PV, Div. 1, Section III, Class MC [92] and Seismic Category I structure according to NRC RG 1.29. It consists of a cylinder, hemispherical dome and torispherical bottom. It rests on a concrete base which is part of the common concrete mat. It is designed to contain the energy and radioactive material that might be released in a postulated LOCA. It provides a high degree of leak-tightness during normal operation and under accident conditions.

Steel containment is briefly described for a better insight into the overall design concept of the Containment at NEK, but it is not part of this review.

The NEK Shield building elements required by the specification include:

- Concrete;
- steel reinforcement;
- the prestressing systems – not applicable for NEK shield building – no prestress systems,
- the liner – not applicable for NEK - shield building contains no liner;
- interaction of the liner with the concrete containment structure such as: anchors to the concrete (e.g. studs, structural steel shapes or other steel products), barrel-to-basemat junction - not applicable for NEK - shield building contains no liner/no interactions;
- waterstops, seals and gaskets and protective coating – not applicable for NEK.

Concrete cylindrical wall and spherical segment dome

The concrete Shield building cylindrical wall extends from the top of the foundation mat to elevation 154,86 meters and has an inside diameter of 35.13 meters, and a wall thickness of 0,76 meters. The concrete spherical segment dome has a radius of 22.59 meters and a thickness of 0,61 meters. The ring girder at the top of the cylindrical wall provides the only lateral support for the dome and transfers vertical load to the cylinder.

Access to the inside of the containment vessel through the shield building is provided by a personal airlock (PAL) and an emergency airlock (EAL) steel structure of approximately 2,92 and 1,52 meters in diameter, respectively. An airlock has the dual interior steel doors, which are provided with seals to minimize air leakage. There is also an equipment access hatch, which is shielded by removable concrete blocks on the exterior side.

The design of electrical and mechanical penetration sleeves is such that differential movement between the Shield building and the containment vessel is allowed. The penetrations are designed in a way that the Shield building leak tightness requirements for the annulus are maintained.

The concrete used has a minimum 28-day cylinder compressive strength of 300kg/cm^2.

The materials and quality control of the concrete construction of the Shield building are defined in the specification "Reinforced Concrete Including Formwork," SP-J200-044687-000. Steel reinforcement is in accordance with the requirements of ASTM A615 [111], grade 60.

The material specification, quality control and construction techniques for reinforcing steel of the shield building are as provided in the following specifications:

- "Placement of Reinforcing Steel" SP-J201-044687-000;
- "Reinforcing Bar Splices" SP-J204-044687-000;
- "Reinforcing Steel Fabrication and Delivery" SP-J500-044687-000.

The moments, shears and axial loads at critical areas are determined using analytical techniques. The strength design concept of ACI 318 is used to determine reinforcing size and location. The general reinforcing pattern used in the cylindrical wall is orthogonal with steel bars arranged vertically and circumferentially on both faces of the wall. The reinforcing pattern for the dome is essentially radial and circumferential with the center section arranged orthogonally for ease of placing.

The design of the Shield building is such that the temperature of the reinforced concrete does not exceed 66°C except for local areas at pipe penetrations where 93°C may be reached.

Foundation mat

The foundation mat beneath the reactor building complex is built on soil ranging from a poorly graded, silty sand mixture to silty or clayey, fine sand at the elevation of approximately 87 meters. This mat is common to the reactor building complex and all surrounding auxiliary buildings.

The design groundwater table is at the elevation of finished plant grade. The design of the foundations includes these groundwater conditions in the postulated loading combinations, as well as the buoyancy of each structure. Waterproofing membranes are provided around the exterior surfaces of the foundations up to finished plant grade. In addition, waterstops are provided at planned construction joints, which occur below finished plant grade.

a) Methods and criteria used for selecting components within the scope of the aging management

Scoping and screening of NEK Systems, Structures and Components (SSC), including Reactor Shield Building is done in accordance with NRC 10 CFR 54 [12] and NEI 95-10 [16]. The scope of SSCs for the extension of plant lifetime is identified based on the 10 CFR 54.4 (see 2.3.1 b).

Reactor Shield Building structure has been evaluated according to GALL table III.A1 [18] and EPRI TR 1002950, Rev. 1 [112]. Applicable aging effects requiring management (AERMs) have been determined based on NEK specific component type – material – environment combinations and past operating experience.

b) Processes/procedures for the identification of aging mechanisms for the different materials and components of the concrete structures

In order to fulfil the requirements of 10 CFR 54, NUREG-1801 [18], Section III.A1 is used for the identification of aging mechanisms for different materials and components of the reactor shield building. However, as specified in the Reference [49], Section 07 »Concrete Containment«, this report discusses the aging mechanisms of the following elements of the Shield building:

- Concrete;
- steel reinforcement.

Implementation of monitoring of structures under 10 CFR 50.65 (the Maintenance Rule) [11] is addressed in the Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.160, Rev. 3 [20], and NUMARC 93-01, Rev. 4A [21]. These two documents provide guidance for the development of licensee-specific programs to monitor the condition of structures and structural components within the scope of the Maintenance Rule with no loss of structure or structural component intended function. Such structures monitoring program is recommended by NUREG-1801, "Generic Aging Lessons Learned" [18] to be used for effective aging management of structures important to safety.

Generally, in the scope of the Structures Monitoring Program both non-metallic and metallic structures are included. Since this review is limited to the concrete Shield building, the metallic structures are out of scope.

The monitoring program for structures consists of periodic visual inspections by personnel qualified to monitor structures and components for applicable aging effects such as those described in the American Concrete Institute Standards (ACI) 349.3R [113], ACI 201.1R [114]. Identified aging effects are evaluated by qualified personnel using criteria derived from industry codes and standards contained in the plant current licensing bases.

All identified aging mechanisms are monitored and controlled by the NEK Structures Monitoring Program. The monitoring of the non-metal part of structures is defined per TD-2L [115] and TD-2N [116] programs in compliance with NEK's overall Aging management program MD-5 [1]. A part of the structures monitoring program is also the TD-2U (Steel structures) [117], but it covers metal structures, and therefore it is out of the scope of this review.

For both documents (TD-2N and TD-2L), a comparison with the requirements from the Revision 2 of NUREG-1801 (GALL report), XI.S6, "Structures Monitoring" [18] was made within NEK work report ESD-TR-05/14, "Aging Management Program Evaluation Report (AMPER)" [19].

The program also includes periodic sampling and testing of ground water and the need to assess the impact of any changes in its chemistry on below grade concrete structures.

No requirement for protective coating is specified for the reactor shield building. In line with external appearance of NEK, the reactor shield building outer surfaces have been painted with no effects on its intended functions.

7.1.2 Aging assessment of concrete structures

a) Aging mechanisms requiring management and identification of their significance

NEK uses NUREG-1801; Generic Aging Lessons Learned (GALL) [18], as a reference to determine the aging mechanisms requiring management. The significance of the aging mechanisms is evaluated during the development and approval process the U.S. NRC conducted before the issuance of the GALL document.

Key standards and guidance used to asses aging assessment of concrete structures include:

- NS-G-2.6 - Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants [118];

- 10 CFR 50.65 – Requirements for Monitoring the Effectiveness of Maintenance at NPPs [11];
- American Concrete Institute, ACI 201.1R 08 - Guide for Conducting a Visual Inspection of Concrete in Service [114];
- American Concrete Institute, ACI 201.2R 92 - Guide to Durable Concrete [119];
- American Concrete Institute, ACI 224R 01 - Control of Cracking of Concrete Structures [120];
- American Concrete Institute, ACI 349.3R 02 - Evaluation of Existing Nuclear Safety Related Concrete Structures [113];
- U.S.NRC RG-1.127, rev.1 - Inspection of water-control structures associated with nuclear power plants [121].

NEK documents that are used to prepare aging assessment of concrete structures are:

- USAR, Updated Safety Analysis Report [10];
- MECL Layout Drawings.

Concrete, steel reinforcement

Aging Effects Requiring Management include:

- Change in Material Properties – ACA;
- Change in Material Properties – CES;
- Cracking – ACA;
- Cracking – CES;
- Cracking – RWA;
- Cracking - Freeze/Thaw;
- Cracking – Settlement;
- Loss of Material – ACA;
- Loss of Material – CES;
- Loss of Material - Freeze/Thaw;
- Reduction in Concrete Anchor Capacity – LCD.

<u>AGING MECHANISMS</u>

ACA - Aggressive Chemical Attack
Concrete, being highly alkaline (pH > 12.5), is degraded by strong acids. Chlorides and sulphates of potassium, sodium, and magnesium may attack concrete, depending on their concentrations in soil/ground water that comes into contact with the concrete. Exposed surfaces of Class 1 structures may be subject to sulfur-based acid-rain degradation. The minimum thresholds causing concrete degradation are 500 ppm chlorides and 1500 ppm sulphates.

CES - Corrosion of Embedded Steel
If the pH of concrete in which steel is embedded is reduced below 11.5 by intrusion of aggressive ions (e.g., chlorides > 500 ppm) in the presence of oxygen, embedded steel may corrode. A reduction in pH may be caused by the leaching of alkaline products through cracks, entry of acidic materials or carbonation. Chlorides may be present in the constituents of the original concrete mix. The severity of the corrosion is affected by the properties and types of cement, aggregates and moisture content.

RWA - Reaction with Aggregate
The presence of reactive alkalis in concrete can lead to subsequent reactions with aggregates that may be present. These alkalis are introduced mainly by cement, but also may come from admixtures, salt-contamination, seawater penetration or solutions of deicing salts. These reactions include alkali-silica reactions, cement-aggregate reactions and aggregate carbonate reactions. These reactions may lead to expansion and cracking.

Freeze/Thaw
Repeated freezing and thawing can cause severe degradation of concrete characterized by scaling, cracking and spalling. The cause is water freezing within the pores of the concrete, creating hydraulic pressure. If unrelieved, this pressure will lead to freeze-thaw degradation. If the temperature cannot be controlled, other factors that enhance the resistance of concrete to freeze-thaw degradation are (a) adequate air content (i.e., within ranges specified in ACI 301-84), (b) low permeability, (c) protection until adequate strength has developed and (d) surface coating applied to frequently wet-dry surfaces.

Settlement
Settlement of a structure may occur due to changes in the site conditions (e.g., water table, etc.). The amount of settlement depends on the foundation material.

LCD - Local Concrete Degradation
Reduction in concrete anchor capacity due to local concrete degradation can result from a service-induced cracking or other concrete aging mechanisms.

<u>ENVIRONMENT</u>

Based on environment the aging mechanisms are divided into internal and external aging mechanisms. Internal aging mechanisms are exposure to air/gas indoors, and external aging mechanisms are those due to exposure to air outdoor or to underground.

Air / Gas - Indoor (lnt)
This environment is the one to which the specified internal or external surface of the component or structure is exposed, the humidity-controlled (i.e., air conditioned) environment. Internal gas environments include dry air or inert, non-reactive gases. This generic term is used only with "Common Miscellaneous Material /Environment", where aging effects are not expected to degrade the ability of a structure or component to perform its intended function for the period of extended operation. The term »gas« does not include those of the fire suppression system.

Air Outdoor (Ext)
The outdoor environment consists of moist, possibly salt-laden atmospheric air, ambient temperatures and humidity, and exposure to weather including precipitation and wind. The component is exposed to air and local weather conditions. A component is considered susceptible to a wetted environment when it is submerged it has the potential to collect water, or is subject to external condensation.

Underground (Ext)
Ground water is subsurface water that can be detected in wells, tunnels or drainage galleries, or that flows naturally to the earth's surface via seeps or springs. Soil is a mixture of organic and inorganic materials produced by the weathering of rock and clay minerals or the decomposition of vegetation. Voids containing air and moisture can occupy 30 to 60 percent of the soil volume. Concrete subjected to a ground water/soil environment can be vulnerable to an increase in porosity and permeability, cracking, loss of material (spalling, scaling) or aggressive chemical attack. Other materials with prolonged exposures to ground water or moist soils are subject to the same aging effects as those systems and components exposed to raw water.

b) Establishment of acceptance criteria related to aging mechanisms

For the concrete structure of the reactor Shield building (concrete and reinforcement installed in the concrete), applicable aging mechanisms were determined according to NUREG -1801 (GALL) [18].

All identified aging mechanisms are monitored and controlled by the NEK Structures Monitoring Program. Monitoring of the non-metal part of the structures monitoring is defined per TD-2L [115] and TD-2N [116] programs in compliance with overall Aging Management Program, MD-5 [1] document.

For both documents (TD-2N [116] and TD-2L [115]), a comparison with the requirements from NUREG-1801 (GALL report) [18] was made within NEK report ESD-TR-05/14, "Aging Management Program

Evaluation Report (AMPER)" [19]. In the context of the comparison, the justification for the rejection of some of the prescribed aging mechanisms in the GALL report (for example, the impact of subsidence and the influence of elevated temperatures) was provided. NEK's Structures monitoring program (TD-2N) includes periodic monitoring of settlements, crack propagation and crack width of all buildings in the scope of AMR. Vertical settlements and crack propagation are measured twice a year and results are compared with results from previous measurements. Up to now there has been no increase in settlements and crack propagation. Therefore this AERM has been screened out from NEK AMR. However, any increase in settlement levels as well as crack width and propagation would be detected and evaluated by the program and actions initiated to eliminate the cause and/or introduce necessary repairs. According to GALL elevated temperature induced reduction of strength and modulus is applicable if indoor air general temperature exceeds 66°C (150°F) or local temperature exceeds 93°C (200°F). Indoor air temperatures of the NEK reactor shield building never exceed 66°C (150°F).

Acceptance criteria for inspection's findings are provided within the American Concrete Institute Standards - ACI 349.3R [113] and ACI 201.1R [114]. In case the acceptance criteria are not met, a corrective action is initiated.

7.1.3 Monitoring, testing, sampling and inspection activities for concrete structures

Concrete and reinforcement of the Shield building

The Structures Monitoring Program is consistent with NUREG-1801, Section XI.S6, "Structures monitoring" [18]. The Structures Monitoring Program manages the aging effects of cracking, loss of material, change in material properties and reduction in concrete anchor capacity due to local concrete degradation. The program relies on periodic visual inspections to monitor the condition of structures, structural elements (including component supports) and miscellaneous structural commodities.

The Structures Monitoring Program in NPP Krško for concrete structures is defined in TD-2L "Program nadzora gradbenih objektov in konstrukcij v NE Krško" (Civil Structures Surveillance Program) [115] and TD-2N "Program tehničnih opazovanj gradbenih objektov in konstrukcij" (Civil Structures Technical Observation Program) [116]. The two programs were developed in accordance with 10 CFR 50, Appendix B; Quality Assurance Criteria for NPP and Fuel Reprocessing Plants [36] and 10 CFR 50.65; "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (Maintenance Rule [11]), and national industry codes.

The structural integrity of the concrete Shield building is determined during the shutdown by a visual inspection of the exposed accessible interior and exterior surfaces of the Shield building. This inspection is intended to verify that there are no visible changes in appearance of the concrete surfaces or signs of other abnormal degradation. Any abnormal degradation of the Shield building detected during inspections is reported to the SNSA in a Special Report.

The type of activity and periodicity for the Shield building concrete observations is defined in the Attachment 2 of the program TD-2N [116] (cracks depth measurement, crack progression measurements, material testing and visual inspection of surfaces). Results of observations are described in the Annual report, where acceptance criteria listed in the TD-2N [116] and TD-2L [115] are considered during the evaluation.

Procedures/instructions for carrying out individual activities (e.g. visual inspection, measurement and monitoring of crack widths, etc.) are prepared by the contractor undertaking the observations. NEK's Quality Assurance Department performs regular audits of external organizations that provide such services.

TD-2L: Program nadzora gradbenih objektov in konstrukcij v NE Krško (Surveillance Program for Buildings and Civil Structures) [115]

The purpose of the program is to define the basic requirements and the rules and responsibilities that need to be considered in the development of maintenance programs of buildings in NEK.

Description of past and current activities

The program deals with maintenance activities in the following areas:

- maintenance works on buildings;
- maintenance of coatings;
- cleaning of production facilities – except radiological controlled area;
- the dimensions of the cross sections of the Sava River on individual sections from the state border to the nuclear power plant (every five years);
- maintaining high water embankments;
- technical monitoring of structures/buildings.

Frequencies

Continuous or periodic control (predictive maintenance) is established, as well as diagnostics of the condition of buildings and structures in order to predict the durability and safety of structures.

The results of predictive maintenance are used to monitor the condition of buildings and structures. In this way, maintenance activities can be carried out before any degradations occur, which would endanger the safety, usability and durability of buildings and structures.

Acceptance criteria

Preventive maintenance of building structures and structures in NEK must include predictive, periodic and planned maintenance actions. The purpose is to maintain building structures and structures in such a state that they:

- meet the safety precautionary requirements;
- ensure the usability and durability of buildings in accordance with specifications for construction and operation specifications;
- allow extension of design life.

TD-2N: Program tehničnih opazovanj gradbenih objektov in konstrukcij (Technical Observation Program of Buildings and Civil Structures) [116]

The purpose of this program is to define the basic requirements, rules and responsibilities of all contractors involved in the implementation of programs of technical observations of buildings in the Krško NPP.

The program ensures monitoring and control of the aging of buildings in the Krško NPP in a way that all the requirements of the Aging Management Program project are satisfied.

The subject of this program is reinforced concrete and other massive structures and buildings including embankments.

Technical observations of construction elements include periodic monitoring of settlements, crack propagation, visual inspection (ACI 201.1 R-08 "*Guide for conducting a Visual Inspection of Concrete in Service*" [114]) and crack width of all buildings and elements in AMP scope.

The scope of the program includes the following aging management activities:

1. Visual inspection of above-grade (accessible, inaccessible) and below-grade (underground) concrete areas (See Element 4 for sample size and justification of areas to be inspected);
2. Groundwater chemistry monitoring program to identify the conditions conducive to below-grade (underground) aging mechanisms:
   - corrosion of embedded steel;
   - chemical attack (chloride and sulphate induced degradation);

3. Geodetic monitoring of deflections/ settlements of concrete structures.

The observations of construction have the status of predictive observations. The program deals with:
- regular periodic observation;
- extraordinary observation.

Exceptional inspections are carried out on selected structures in the emergency situations and events:
- after an earthquake intensity VI or more on Mercalli intensity scale;
- after an emergence due to other major natural disasters;
- after major accidents inside or outside the plant;
- before and after major interventions in buildings, e.g. due to reconstruction,
- after identifying any irregularities or damage to buildings or construction elements that could jeopardize the safety of the building or structural element.

The main activities for regular periodic observation and/or extraordinary observation include:
- visual inspection of the condition of concrete surfaces or protective coatings if the concrete surfaces are painted with protective paint;
- measurements of vertical displacements observation rappers;
- measurements of expansion joints between objects;
- cracks depth measurements;
- periodic testing of materials and laboratory tests on samples;
- laboratory investigation of groundwater to identify conditions below-grade, if there is any chemical attack (chloride and sulphate degradation) on concrete.

The program provides means to address the following aging effects and mechanisms, as described in ACI 349.3R-02 [113] and ASCE/SEI 11 -99 [122]:

- cracking or loss of material (spalling, scaling) due to freeze-thaw degradation;
- cracking or loss of material (spalling, scaling) due to chemical attack (chloride, sulphate induced);
- cracking and loss of strength due to cement aggregate reactions;
- cracking, loss of material and loss of bond due to corrosion of embedded steel;
- increase in porosity/permeability and loss of strength due to leaching of calcium hydroxide ($Ca(OH)_2$);
- cracking and distortion due to long-term settlement.

Frequencies

The evaluation frequency is based on the aggressiveness of environmental impacts as well as physical conditions of plant structures. The frequency is established in a way to ensure that any age-related degradation is detected at an early stage.

The proposed inspection schedule for visual inspection is defined in ACI 349.3R-02 [113]:

- Below-grade structures < 10 years;
- Structures exposed to natural environment (direct and indirect) < 5 years.

Vertical settlements and crack propagation are measured twice a year (winter and summer period to monitor the effect of temperature in two extreme climatic conditions), and results are compared to results from previous measurements. Visual inspection – visual check on the condition of concrete surface is done once a year which is more than is required by the ACI 349.3R-02 [113] standard.

Acceptance criteria

The results of observations are compared against the following criteria:

- Absolute values of vertical and horizontal displacements: the measured movements are within the limits of the order of accuracy of 1-2 mm;
- relative values of vertical and horizontal movements: the same as absolute values;
- the width of the cracks in the concrete: for the previous identified cracks, repair is necessary at a width of grater then 0.2 mm;
- the depth of the cracks in the concrete: for the previously identified cracks, the depth should not increase in the case of an increase, the causes should be analyzed;
- carbonization of concrete (pH > 11.5),
- the presence of chlorides and sulphates in groundwater (chlorides < 500 ppm and sulphate < 1500 ppm);
- damage and impact of damage on the function of the structure.

7.1.4 Preventive and remedial actions for concrete structures

Preventive actions include continued inspections and monitoring according to the plant Structures Monitoring Program (TD-2N [116]) to ensure that design limits are not exceeded. These inspections are part of the approved design basis and will be continued for the sample size and inspection frequency, as established in the program TD-2N.

Preventive actions further include painting the concrete (and steel) surfaces to protect against the impact from outer and internal environment.

Remedial actions are repairs of concrete according to the ACI 349.3R-02 chapter 8 [113].

Objectives of Remedial Work Include:

- Restore component structural integrity;
- Arrest mechanisms producing distress;
- Ensure (as far as possible) that cause(s) of distress will not reoccur.

The acceptance criteria against which the need for corrective actions is evaluated, ensures that the structure and component intended function(s) are maintained under all CLB design conditions during the period of extended operation. NEK has in place a Corrective Actions Program [38], quality assurance (QA) procedures, site review and approval process and administrative controls are implemented in accordance with 10 CFR Part 50, Appendix B

7.2 Licensee's experience of the application of AMPs for concrete structures

The Structures monitoring program has been implemented at NEK. Records of visual inspections from the beginning of the plant operation are available. Visual inspections in accordance with ACI 201.1 R-08 standard [114] are performed since 2016.

During inspection, some indications on concrete were detected. These degradations were attributed to drying shrinkage and freeze-thaw cycle. None of those were relevant to the extent to require repair actions to be taken. All non-relevant indications identified in inspections will be monitored in future inspections.

The AMP for concrete structures evaluates applicable operating experience, both past and current and will continue to do so as new operating experience is accumulated after the license renewal, including:

- internal and industry wide condition reports;
- internal and industry wide corrective action reports;
- vendor-issued safety bulletins;

- NRC Generic Communications;
- applicable DOE or industry initiatives (e.g., EPRI- or DOE-sponsored inspections).

The AMP goal is to clearly identify any degradation as determined in the operating experience, as either age-related or event-driven and establish appropriate justification for such an assessment. In this way past operating experience supports the adequacy of the proposed AMP including the method/techniques, acceptance criteria as well as the frequency of inspection.

7.3 Regulator's assessment and conclusions on ageing management of concrete structures

a) Assessment of the ageing management processes

The Krško NPP concrete shield building ageing is managed with the program TD-2N "Program tehničnih opazovanj gradbenih objektov in konstrukcij" (Civil Structures Technical Observations Progam). Within the program the instructions are provided for carrying out the regular periodic and extraordinary inspections of concrete structures. Concrete structures of Seismic category I, structures required for functional and safe operation, objects with complex construction structures, and objects identified by AMP are included in the program scope. The program includes requirements from GALL section XI. S6 (Structures Monitoring Program).

The containment shield building inspection is performed on inner and outer cylindrical wall surfaces as well as on containment inner and outer dome surfaces. The inspection period for inner surfaces is 18 months and for external surfaces is 6 months (visually with binoculars).

The containment shield building inspection is carried out by the contracted external organizations. The Krško NPP staff review and approve contractors' implementation procedures. In the TD-2N program, the basic requirements for the content of technical implementation procedures are provided (includes: technical implementation of observations and measurements, measurement and other equipment, expected measurements accuracy, method of documenting observations and measurements and the evaluation of results of the overall process).

Possible irregularities detected by the visual inspection of the containment shield building are evaluated and recorded in the corrective actions program (CAP). The detailed inspection is then carried out and the method of remediation is determined. Cracks measurement and their evaluation according to the defined acceptance criteria are carried out in accordance with ACI standards. Identified cracks are recorded and periodically monitored (trending).

b) Experience from inspection and assessment as part of its regulatory oversight

The ageing management program implementation of plant concrete structures is regulatory oversighted during regular or thematic inspections. The concrete containment shield building was also covered by an extensive five day SNSA inspection.
The SNSA inspection found out that program TD-2N of concrete structures is consistent with attributes from GALL - *XI. S6 Structures Monitoring Program*, and requirements from GALL *Chapter III - Structures and component supports* for PWR Shield Building - concrete part of the containment structure. According to the criteria from NUREG-1801, the NEK carried out the selection and evaluation of concrete structures (scoping/screening), and AMR reports were prepared including the ageing management requirements. Based on the AMR a revision of the TD-2N program was made.
The Krško NPP also explained the basis for the main requirements of the program, such as the frequency of visual inspections of concrete structures (twice a year) and the requirements for extraordinary inspections of concrete structures including concrete containment shield building. The required periodicity according to ACI standards is 5 years. Therefore, a choice of two inspections per year is a conservative approach. Extraordinary inspections are carried out in case of emergency situations and events with potential for

structure damage (e.g. earthquake, natural disasters, operational accidents, interventions in structures, identified anomalies or damage to structures).

The basic acceptance criteria are taken from the ACI standards. Certain internal NEK criteria are more stringent than standards from industry practice.

The Krško NPP also started systematically with the material sampling of concrete structures in 2016. The required tests are performed in an accredited laboratory by the contractor. Reactor building concrete samples are taken on occasion when maintenance work or modifications are being implemented, involving the concrete structures.

c) The main strengths and weaknesses that have been identified either by the licensees or the regulator on the effectiveness of the SSC specific AMPs

Main strengths:

- The program TD-2N covers systematically all aspects of ageing management of concrete structures in accordance with GALL requirements;
- Certain internal NEK criteria used in the ageing management program are even more stringent than industry standards given criteria. The Krško NPP also carries out systematic material sampling for concrete structures material testing.

Main weaknesses:

- It is recognized that the Krško NPP should improve the coordination and complete overview over the work of external contracted organizations. Generally, the Krško NPP reviews all implementation procedures of the contractor, and also the results of their activities, but does not always have enough time and resources to examine and supervise their work in detail.

d) The conclusions on the adequacy of the licensee's SSC specific ageing management programme(s)

Programs and procedures for the concrete containment shield building ageing management implementation have proven to be appropriate and effective. The Krško NPP has no negative experiences that would require changes of the methods or acceptance criteria used in the program. All previous inspections of the concrete shield building, carried out according to the TD-2N program have been performed to the intended extent (examinations of the outer and inner surfaces of the cylindrical wall and dome). There was no unexpected degradation found. The program TD-2N is consistent with the requirements given in GALL - *XI.S6 Structures Monitoring Program* section.

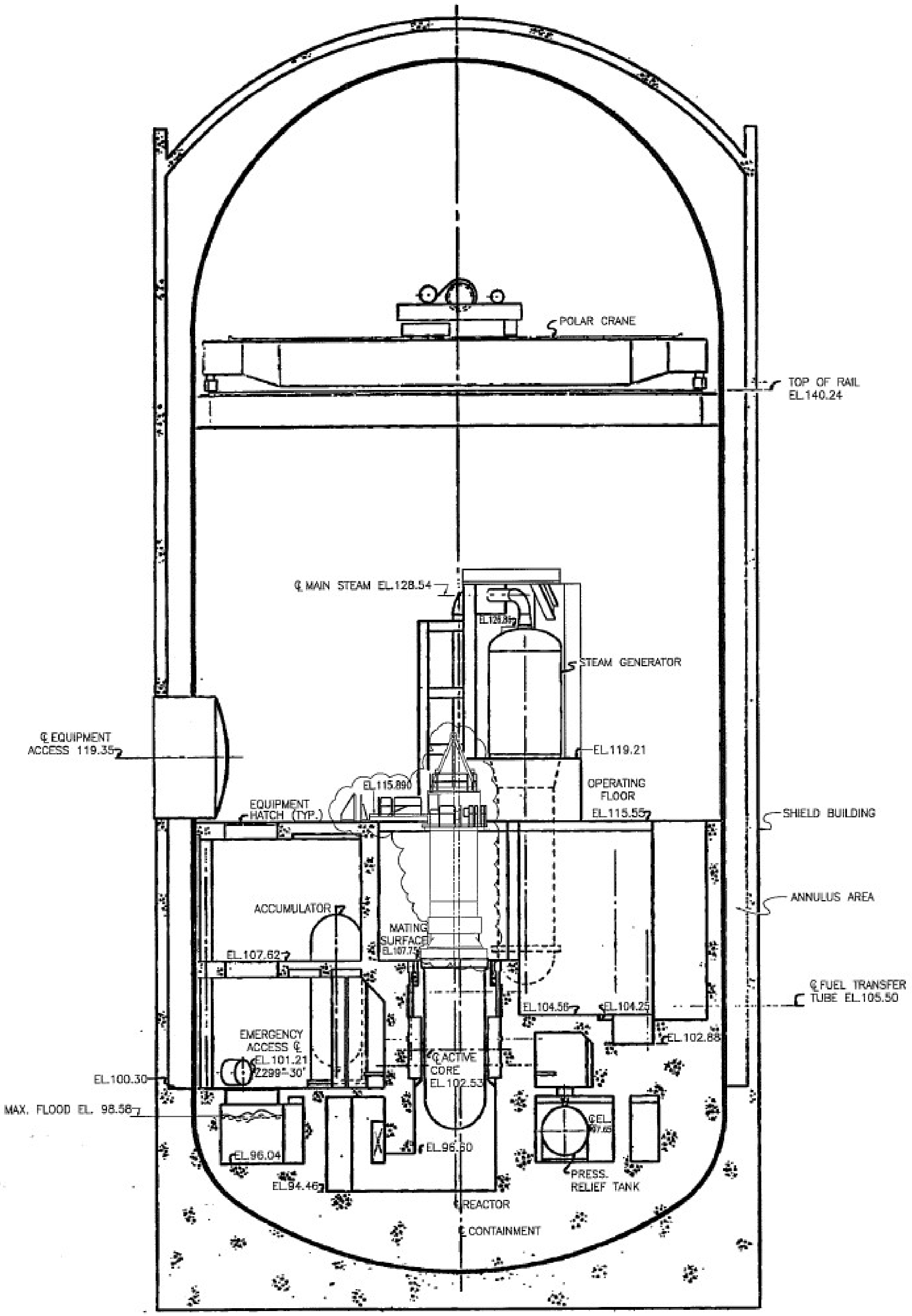

Figure 7-1: Reactor Building Layout

8 Pre-stressed concrete pressure vessels (AGR)

Not applicable for the Krško NPP.

9 Overall assessment and general conclusions

SNSA participated in the preparation phase of technical specifications and very early informed the Krško NPP about TPR process. After the start of the process, a decision has been issued to the Krško NPP for the TPR on ageing management and for preparation of the Technical report. SNSA afterwards carried out extensive inspection covering all thematic areas from the TPR. The Krško NPP prepared Technical report with attached expert opinion from the TSO. The SNSA than reviewed the whole report and added general part and chapters related to administrative control and evaluation of the AMP to prepare a national report. As mentioned in the report, the Krško NPP intends to continue operation beyond its original design life to 60 years and prepared all necessary technical documents to justify the plant's life extension during the Aging management review project. Based on results of the project, the Krško NPP established AMP documented in MD-5 Aging Management Program, which was issued in 2010. Based on the regulatory review and review of international group of experts, the SNSA decided to approve AMP as one of the prerequisites for possible lifetime extension from 40 to 60 years of operation.

Aging management approach in the Krško NPP for the preparation of LTO and license renewal activities are compliant with U.S. NRC regulations and industry practices. Implemented aging management processes fully meet the requirements of NUREG-1801 – GALL [18] and the Slovenian legislation. Namely, "Ionizing Radiation Protection and Nuclear Safety Act (ZVISJV)" [5] and "Rules on Operational Safety of Radiation or Nuclear Facilities (JV9)" [6], where requirements for aging management are given. Ageing management methodology is in general similar to the IAEA standards and guidelines for ageing management and was confirmed with conclusions of the Operating Safety Review Team Mission(OSART). The mission carried out in the Krško NPP in 2017 and considered Long Term Operation. Besides that, ageing management for active components is provided in the Maintenance rule.

The Krško NPP developed and implemented appropriate programs and procedures including methods for identification and monitoring of the effects of ageing and requirements for the implementation of preventive and corrective measures. The Krško NPP also reports to the regulator annually on experience and results of the implementation of the ageing management programs in accordance with JV9. Aging management program in the Krško NPP is reviewed in a five-year period to implement all recommended changes of processes, procedures and programs. The Krško NPP has made additional improvements to individual ageing management programs, also based on the evaluation of changes, derived from the second revision of GALL. After several thematic inspections and other regulatory reviews the SNSA also concluded that generally all appropriate programs and procedures for ageing management in the Krško NPP are developed and implemented and so far no unacceptable degradations have been identified that would affect the safe operation of the power plant. Ageing management programs are completed and proven to be in accordance with GALL requirements and their implementation is in a very comprehensive phase much earlier before the end of designed operational lifetime. Ownership responsibility for ageing management activities is on a very high level and the staff have in general a very proactive attitude. Beside that the Krško NPP has some remaining work to do, since not all technical implementing procedures deriving from ageing management programs have been implemented yet.

During the implementation of the cable aging management program, the Krško NPP found some localized "Hot Spots", where cable jacket showed the effects of thermal degradation. Nevertheless, the primary insulation was found to be in acceptable condition. The Krško NPP concluded the first cycle of required aging management inspections for MV cables (started 2010) and initiated the second cycle, where the focus is on trending of the results from the first cycle. All activities in accordance with GALL [18] requirements will be concluded before transition to extended plant life time in 2023. The aging management activities will then continue until the final shutdown for decommission. The activities of the Cable Aging Management Program provide reasonable assurance that the intended functions of electrical cables and connections exposed to the adverse environments will be maintained consistent with the current licensing basis through the period of extended operation. The program is well developed, but it is important to finalize procedures for diagnostic testing of electrical cables and to revise criteria for sample selection in case of degradation finding. With new criteria for sample selection the sampling list also needs to be updated.

During the implementation of the aging management program for concealed pipework, the Krško NPP did not find any relevant indications, which would require actions to be taken. All inspected pipelines and tanks are in a good condition. Several minor corrosion indications were found on the SW discharge pipelines, which will be monitored in future inspections to assure there is no deterioration in the future. Considering results from concealed pipework inspections, there was no need for any changes in the applicable aging management program. The Krško NPP concluded the first cycle of concealed pipework aging management inspections and is preparing for the second cycle. The second cycle will be focused on the trending of the results from the first cycle, especially on those pipeline sections where indications of corrosion have been found.

During the implementation of aging management program for the reactor pressure vessel, the Krško NPP also did not find any relevant indications which would require actions to be taken. The reactor pressure vessel fulfills the acceptance criteria regarding irradiation embrittlement, mechanical fatigue and PWSCC. The reactor pressure vessel head was replaced in 2013 in order to exclude the risk of primary water stress corrosion cracking. For the Reactor Vessel components, no indications of eventual propagation of aging related degradation was found. The SNSA also considers it important, that the possibility of hydrogen flaking in the base material of reactor pressure vessel was analyzed in the Krško NPP, even if it is not proven to be an issue related to ageing. Hydrogen flaking is as concluded, not expected in the Krško NPP according to available information on the technological process of the reactor pressure vessel manufacturing, the heat treatment and the results of the pre-service inspection. Additionally, the Krško NPP carried out preventive UT inspection of the reactor pressure vessel calibration blocks and no indications were detected.

During the implementation of aging management programs for the shield building of the primary containment, the Krško NPP did not find any relevant indications, which would require corrective actions to be implemented. All inspected concrete shows to be in very good condition. Several minor non-relevant indications were found on the shield building wall, which will be monitored (trended) in future inspections. Considering the results from the shield building inspections, there was no need for changes to the aging management program of structures. Programs and procedures for ageing management implementation have proven to be appropriate and effective. Certain internal criteria in the Krško NPP used in the ageing management program are even more stringent than criteria from industry practice. The Krško NPP also carries out systematic material sampling for concrete structures material testing. On the other hand it is recognized that in some cases the Krško NPP should improve the coordination and overview of the work of external contracted organizations, since there has not always been enough time and resources to examine and supervise their work in detail.

The Krško NPP aging management program is a living program with inherent strength for improvements based on internal and external operating experiences and results of R&D worldwide. The Krško NPP ageing management program as described also fully complies with the requirements and expectations set by the Slovenian regulation [5], [6] and international best practices. In spite of the above mentioned revealed findings identified during TPR, which are relatively easy to resolve by the Krško NPP, there are also several challenges and areas for improvement in the future. Ageing management of electrical cables is the most developing and interesting area for R&D programs and activities. The Krško NPP is already now preparing its own programs regarding cables area and corporates with different external institutes and laboratories. However, there is still much to do in this field. Influence of pressure vessel irradiation on the Krško NPP lifetime is also an important area for R&D in the future. A Systematic monitoring and addressing foreign operational experience in the field of ageing is also one of the key elements for safe long-term operation. Current issues in other countries, such as hydrogen flaking in the reactor pressure vessel material, which could affect the lifetime of the power plant should not be overlooked. Besides that, the power plants have to implement appropriate preventive measures related to ageing management, such as monitoring of cable spreading areas and testing of cables, coating and cathodic protection of buried piping, replacement of components more resistant to PWSCC and preventive inspections and monitoring of concrete structures. Most of which have already been introduced in the Krško NPP.

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