NATIONAL ASSESSMENT REPORT OF THE SLOVAK REPUBLIC

for the Purposes of Topical Peer Review on "Ageing Management" under the Nuclear Safety Directive 2014/87/EURATOM

December 2017

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Abbreviations used

AM	Ageing management
AMP	Ageing management program
ASME	American Society of Mechanical Engineers
BM	Base metal
BT	Safety class
CA	Control assembly
CADAK	Cable ageing data and knowledge
CCW	Circulating cooling water
CODAP	Component operational experience, Degradation and ageing programme
CPS	Central pumping station
DATD	Digital archive of technical documentation
DGS	Diesel generator station
DRS	Ageing management databank
EBO V2	Bohunice NPP
ECCS	Emergency core cooling system
EMO	Mochovce (units 1&2) NPP
ENSREG	European Nuclear Safety Regulators Group
EPR	Ethylene propylene rubber
EPRI	Electric Power Research Institute
EQR	Equipment reliability
ESW	Essential service water
EU	European Union
FCT	Fan cooling tower
GO	General outage
HAZ	Heat affected zone
HELB	High energy line break
HPCS	High-pressure compressor station
HVAC	Heat ventilation and air conditioning system
IAEA	International Atomic Energy Agency
IGALL	International Generic Ageing Lessons Learned
INPO	Institute of Nuclear Power Operations
IQAP	Individual quality assurance program
IMS	Integrated quality management system
I&C	Instrumentation and control system
LCO	Limits & Conditions
LOCA	Loss of coolant accident
LTO	Long-term operation
MCP	Main circulation pump
MO3&4	Mochovce (units 3&4) NPP (under construction)
MNA	Methodical instruction
MSH	Main steam header
MT	Maintenance template
NAR	National assessment report
NA	Guide
NDT	Non-destructive test
NESW	Non-essential service water
NFM	Neutron flux monitoring
NG	Notification General

NI	Nuclear installation
NPP	Nuclear power plant
OIT(p)	Oxidation induction time/ temperature
PC	Primary circuit
PE	Polyethylene
PRZ	Pressurizer
PSA	Probabilistic safety assessment
PSR	Periodic safety review
PTS	Pressurized thermal shock
PVC	Poly-vinyl-chloride
QP	Quality plan
RHWG	Reactor Harmonisation Working Group
RPV	Reactor pressure vessel
RSA	Release station to atmosphere
SALTO	Safety aspects of long-term operation
SAP	SAP software
SC	Secondary circuit
SE, a. s.	Slovenské elektrárne, a. s. (the holder of the licence for operation of NPP)
SEPS	Super-emergency power supply
SG	Steam generator
SSC	System, structure, component
SMS	System maintenance strategy
SNaP	Correction and prevention system
SPT	Small punch test
SR	Slovak Republic
STaPD	Technical and design documentation centre
STD	Accompanying technical documentation
STN	Slovak technical standard
SUP	Surveillance programs
TAM	Thermal ageing monitoring
TDR	Time domain reflectometry
TK	Technological channel
TLAA	Time limited ageing analysis
TPP	Technological operating procedure
TPR	Topical peer review
ÚJD SR	Úrad jadrového dozoru Slovenskej republiky/ Nuclear Regulatory Authority of the Slovak Republic
WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulators Association
WWER	Water-cooled and water-moderated energetic reactor
WM	Weld metal
XPE	Cross-linked polyethylene

Preamble

Slovakia is a country with more than 60 years of experience in the construction and operation of nuclear power plants (NPPs). Currently there are four units of WWER 440/V213 type operated in Slovakia – two in Bohunice and two in Mochovce. In Mochovce site there are also two units with WWER 440/V213 under construction. The total installed capacity of the units in operation is 1,950 MWe. Three other nuclear units in Bohunice site are under decommissioning – the first Czecho-Slovak unit, A1 cooled by gas and moderated by heavy water, and two units of older WWER 440/V230 type.

The owner and holder of licence for operation of all nuclear units in operation and units under construction is the joint stock company Slovenské elektrárne, a. s. (SE, a. s.). The state regulatory authority for state supervision over nuclear safety of nuclear installations is the Nuclear Regulatory Authority of the Slovak Republic (ÚJD SR).

Council Directive 2014/87/EURATOM of 8 July 2014 amending the Directive 2009/71/EURATOM, establishing the Community framework for the nuclear safety of nuclear installations of the European Union requires that the Member States of the European Union shall conduct Topical Peer Review – TPR every six years with the first review being in 2017. The objective of the Topical Peer Review process, as agreed by ENSREG, is:

- To enable participating countries to review their measures for ageing management of nuclear power plants, to identify good practice and areas for improvement,
- To conduct European peer review, to share operational experience and to identify common issues faced by the EU Member States,
- To provide an open and transparent framework for participating countries, to identify appropriate follow-up measures to remedy areas for improvement.

The EU Member States, acting through ENSREG, agreed that the topic for the first Topical Peer Review will be ageing management. The task is further specified and its scope is outlined in the document "Report Topical Peer Review 2017 – Ageing Management, Technical Specification for the National Assessment Reports", developed by WENRA and approved by ENSREG.

Based on the Council Directive 2014/87/EURATOM and Atomic Act (Act No. 541/2004 Coll.), the licence holder is obliged to carry out a TPR based on UJD SR's specification. The subject of 2016-2017 TPR is ageing management of the following nuclear installations:

- Nuclear power plants,
- Research reactors with a power equal to 1 MW_{th} or more.

Research reactors with a power below 1 MW_{th} may also be included on a voluntary basis.

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

- Operating on 31st December 2017, or
- Under construction on 31st December 2016.

Those installations, which are permanently shut down and have a regulatory or competent authority obligation not to operate or generate electricity beyond 31st December 2017 are excluded from the National Assessment Report.

Nuclear installations under constructions are those for which permission for construction has been granted. For these it is expected that ageing phenomena will have been considered in design assumptions. Thus, some detailed parts of this specification would generally not be applicable at the current stage of their lifetime.

Subsequently, Slovakia has conducted the required assessment in the field of ageing management on the NPPs in operation and under construction, and elaborated the National Assessment Report.

This National Assessment Report describes the used methodology, procedure and results obtained from the ageing management assessment of 6 WWER units belonging to three nuclear power plants in Slovakia: Bohunice V2 (EBO V2) and Mochovce units 1&2 (EMO), which are in operation and Mochovce units 3&4 (MO3&4) under construction. The report was prepared by ÚJD SR in close co-operation with SE, a. s., on the basis of Slovak legislation and documents provided by SE, a. s.

Summary

In Slovakia, there are currently four nuclear units WWER 440/V213 type in operation, two units in Bohunice site and two units in Mochovce site. Besides, there are two additional WWER 440/V213 units in Mochovce under construction. The total installed capacity of units in operation is 1,950 MWe. The owner and holder of licence for operation of all units in operation in Slovakia is Slovenské elektrárne, a. s. (SE, a. s.).

State supervision over the nuclear safety of nuclear installations is exercised by the Nuclear Regulatory Authority of the Slovak Republic (ÚJD SR). State supervision is conducted under the Atomic Act (Act No. 541/2004 Coll.) and the related set of decrees, in particular ÚJD SR Decree No. 430/2011 Coll. on the requirements for nuclear safety, and the ÚJD SR Decree No. 33/2012 Coll. amended by Decree 106/2016 Coll. on periodic, comprehensive and systematic nuclear safety review. The entire body of this legislative framework is regularly updated in line with the IAEA safety standards and WENRA reference levels [2].

All nuclear power plants have their safety reports that are updated according to the regulator´s requirements and reviewed by the regulator. In accordance with the current national legislation, the SAR update on nuclear installations in Slovakia is currently being implemented continuously. Existing Probabilistic Safety Assessment studies (PSA Level 1 and Level 2) confirm that the nuclear power plants meet internationally recognized safety targets. The Probabilistic Safety Assessment studies (PSA Level 1 and Level 2) are regularly updated. The latest update for NPP EBO V2 was in 2015 and for NPP EMO in 2016. Based on the results from the EU stress tests, some specific parts of the safety documentation related to the assessment of rare extreme external hazards and the implementation of severe accident management measures were updated.

In accordance with the requirements of the national legislation, all nuclear power plants in Slovakia are subject to a periodic safety review carried out at regular 10 year interval. The last periodic safety review for NPP EBO V2 was in 2008 (new periodic review for NPP EBO V2 is ongoing), for NPP EMO in 2011 (new periodic review for NPP EMO is ongoing). Based on the results of reviews, ÚJD SR approves programs of further safety improvements aimed at achieving even closer compliance with the current safety standards. Approved programs also include implementation of comprehensive measures to mitigate consequences of severe accidents.

All operated units in Slovakia were subjected to independent reviews of many international missions. Since 1991, there had been more than 20 IAEA missions in total (site review, design review, OSART and IPSART missions), several WANO missions, two RISKAUDIT missions and one WENRA mission.

In 2003 the IAEA initiated an extra budgetary program entitled SALTO (Safety Aspects of Long-Term Operation of Water Moderated Reactors). In addition to unifying and optimising approaches when permitting long-term operation (LTO) the IAEA also followed another objective, which was to provide guidance to member states where the process of licensing long-term operation is expected.

Based on results and conclusions of SALTO program the IAEA released the safety guide SRS-57 “Safe long-term operation of nuclear power plants”.

Safety guide SRS-57 “Safe long-term operation of nuclear power plants” provides information on good engineering practices, which can be referred to when developing the national programs of long-term operation of nuclear power plants. It provides the operator and the regulatory bodies with

guidelines for demonstrating and verification of safety of nuclear power plants. This safety guide was used as a basis in preparation of guidelines for international peer reviews focusing on safety of the long-term operation “Guidelines for peer review of long-term operation and ageing management of Nuclear Power Plants”.

At the request of the government of the Slovak Republic an IAEA Operational Safety Review Team (OSART) visited NPP EBO V2 in 2010. The purpose of the mission was to review operating practices in the area of management organisation and administration – operations, maintenance, technical support, radiation protections, operating experience, chemistry and emergency planning and preparedness. At the request of the plant the team also reviewed the Long-Term Operation programs. In addition, an exchange of technical experience and knowledge took place between the experts and their plant counterparts on how the common goal of excellence in operational safety could be further pursued.

In connection with the long-term operation program, the SSC ageing management programs are developed for a period of 60 years of operation, which the WANO PEER REVIEW mission at the NPP EMO in operation conducted in 2013, and OSART mission (extended with the LTO module) on NPP EBO V2 in operation conducted in 2010, classified as “Good Practice“.

In particular, the surveillance specimen program at NPP EBO V2 was extended to include new materials located in the reactor core, and covers operational conditions under increased unit power, and use of new type of nuclear fuel for a period of 60 years. The OSART 2010 mission at NPP EBO V2 identified this program as “Good Practice“.

This National Assessment Report (NAR) includes an assessment of the ageing management process at NIs in Slovakia for the purposes of Topical Peer Review, to be carried out in 2018 according to Council Directive 2014/87/EURATOM of 8 July 2014 amending the Directive 2009/71/EURATOM, establishing the Community framework for the nuclear safety of nuclear installations of the European Union. The structure and the content of the NAR were prepared according to the ENSREG specification. The NAR has been developed by ÚJD SR in close co-operation with SE, a. s., based on Slovak legislation and documents supplied by SE, a. s. The subject of assessment are nuclear power plants in operation and under construction in Slovakia (Chapter 1.1). Currently only NPPs of WWER 440/V213 type are operated or being built in Slovakia, and the owner/holder of licence for operation of all units in Slovakia is only Slovenské elektrárne, a. s. (SE, a. s.). However, the assessment is also documented for a specific type of NPP indicating the differences between the individual NPPs, as long as these differences are significant and important for the ageing management assessment. Examples of the implementation of the overall AMP in practice are presented for selected equipment, including power cables, concealed pipes, reactor pressure vessel and concrete containment structures.

Ageing management and life cycle assessment of NPPs have been implemented since 1991, while ageing management has been part of several projects aimed at increasing nuclear safety and reliability of NPP operation.

The rules of systematic approach to SSC ageing management are legislatively defined in several ÚJD SR documents. These documents are based for example on the recommendations of the IAEA Requirements for Commissioning and Operation of NPPs [9], Safety Guide for Ageing Management [1] and WENRA [2]. Ageing management is one of the areas under review in the periodic safety review of nuclear installations.

The basic legislative requirements are reflected in the process documentation of the licensees' Integrated Quality Management System (IMS) and in the relevant ageing management programs developed for SSC relevant for nuclear safety. The licence holder has a proactive ageing management system (i.e. with foresight and anticipation) for SSC, which are relevant for nuclear safety with the aim to maintain their design safety functions during long-term operation. The ageing management process has been implemented on the units of NPP EBO V2, NPP EMO in operation, as well as on NPP units MO3&4 under construction.

For implementation of ageing management, SE, a. s., has a Group for Lifecycle Management – Nuclear. The ageing management is included in the process model of the licence holder in the top process of Generation, Process Engineering.

The Ageing Management Program (AMP) for cables is implemented by the licence holder and is performed in accordance with the guide – Cables Ageing Management Program. This guide is valid for all nuclear units in Slovakia, i.e. for the NPP EBO V2, NPP EMO in operation and units of NPP MO3&4 under construction. By individual sub-programs within the cables AMP (surveillance specimen program, measurements of functional cables in operation), the licence holder covers the main degradation mechanisms identified on the basis of operational experience and international recommendations. The licence holder also monitors ambient parameters (temperature, radiation dose, relative humidity), to which the cables are exposed during operation. Monitoring includes both containment and areas outside the containment at both nuclear power plants in operation.

Ageing management of concealed pipes is part of AMP for the pipes of essential service water system – Ageing Management Program for the Pipes of Essential Service Water System. This guide is valid for NPP EBO V2 and NPP EMO in operation. For units of NPP MO3&4 in the construction phase, the AMP will become effective before their commissioning. The scope of activities within AMP of essential service water system (corrosion monitoring, concrete monolith monitoring, wall thickness measurement, visual inspections) covers the monitoring of all relevant degradation mechanisms identified on the basis of operational experience, international recommendations and results of the ageing management program. Based on monitoring of the conditions of the essential service water pipes at the NPP EBO V2, reconstruction and replacement of these pipes was realized.

Ageing management program of the reactor pressure vessel is implemented by the licence holder and is performed in accordance with the guide – Ageing Management Program for the Reactor Pressure Vessel. This guide is valid for all nuclear units in Slovakia, i.e. for the NPP EBO V2, NPP EMO in operation and units of NPP MO3&4 under construction. The scope of AMP for the RPV activities (surveillance specimen program, fluency monitoring, fatigue assessment, in-service inspections) covers monitoring of all relevant degradation mechanisms identified based on operational experience, international recommendations and results of ageing management program. The surveillance specimen program has been extended with new materials located in the reactor core. The program covers operating conditions for increased power of nuclear units and for the use of new type of nuclear fuel.

The licence holder, in accordance with the Decision [3], sends to ÚJD SR regular reports on the results of ageing management programs, focusing on lifetime of the reactor pressure vessel and classified equipment of the unit, including the critical temperature of the RPV brittle fracture, evaluation of the critical temperature of brittleness of RPV fracture, based on tests of chain of surveillance RPV specimens and evaluation of in-service inspection programs.

Ageing management program for the nuclear power plant containment is implemented by the licence holder and is performed in accordance with the guide – Ageing Management Program for the Main Generating Unit. This guide is valid for NPP EBO V2 and NPP EMO in operation. For the NPP units MO3&4 under the construction, the containment AMP will become effective before their commissioning. Since the NPP units were put into operation, periodic monitoring of tightness and containment strength is performed, as well as geodetic measurement and evaluation of settlement of the main generating unit.

The licence holder is involved in the following international projects: IAEA IGALL, OECD/NEA CADAK, OECD/NEA CODAP and OECD HALDEN. The licence holder has access to the databases and materials of the Electric Power Research Institute (EPRI) in the field of ageing management. The licence holder is a member of the International Equipment Reliability Working Group focusing on exchange of experiences in the Equipment Reliability process.

The assessment carried out confirmed that Slovakia has established a legislative basis for ageing management. Ageing management at the nuclear power plants in Slovakia is ensured in a systemic way and can be considered as mature. The licence holder has an established ageing management program to identify all ageing mechanisms related to systems, structures and components (SSC) relevant for nuclear safety. Under this ageing management program the licence holder analyses, monitors and documents SSC degradation as a result of ageing. It defines the possible consequences of ageing, and sets out the necessary activities to maintain the operational capacity and reliability of SSC.

Regular inspections are carried out by ÚJD SR at the licensee to check the compliance with the legal requirements and international safety standards and best practices. The licensee is encouraged to further implement the AMP by participation in the international projects and to exchange experience with other NPP operators.

During the preparation of the national report the following good practices have been identified:
- In the strategic plans in the field of AM, in the methodology documentation the licence holder addresses not only the issue of obsolescence, but also the development of long-term rehabilitation strategy for SSCs,
- The licence holder maintains a special database for the AM purposes,
- Development of AMP for classified equipment for a long-term operation.

The process also identified challenges:
- Deficiencies in the SSC drawings in relation to the actual status,
- Non-continual updating of AM database to reflect the actual status of SSCs and knowledge.

Capability of systems, structures and components relevant in terms of nuclear safety to fulfil their safety functions is considered by ÚJD SR as secured.

1 General information

In Slovakia, there are four WWER 440/V213 nuclear units in operation, two units in Bohunice site and two units in Mochovce site. In addition, there are another two WWER 440/V213 units under construction in Mochovce with significantly improved design. The total installed capacity of all units in operation is 1,950 MWe. The owner and the holder of operating licence for all these units in Slovakia is the company – Slovenské elektrárne, a. s. (SE, a. s.).

1.1 Identification of nuclear installations

Slovakia is an inland country situated in a temperate climate zone in Central Europe. There are two nuclear sites in Slovakia: Bohunice with two operated units of NPP EBO V2 and Mochovce with two operated units of NPP EMO and two other units under construction – NPP MO3&4, which altogether form NPP Mochovce (see the location of sites on the map and view of individual sites on Figures 1-1, 1-2 and 1-3).

The licence holder for all these units is the joint stock company, Slovenské elektrárne, a. s., with its seat at: Mlynské nivy 47, 821 09 Bratislava.

Figure 1-1 Location of NPPs in Slovakia

Bohunice site is located in western Slovakia; the nearest towns are Trnava, Hlohovec and Piešťany. Cooling water is fed from the Váh river, located approximately 8 km east of the site; the difference in the altitude is more than 20 m. On the Váh river there is a reservoir Sĺňava with a water surface of approximately 480 ha and a maximum water volume of 12.3 million m^3. The service water from the Sĺňava reservoir is delivered to NPP EBO V2 through a pumping station. Offtake of service water from the Sĺňava reservoir is through four siphon pipes to the Drahovce pumping station, from there the water flows through four pipes gravitationally through gate valves into the suction pit of Pečeňady pumping station. From the pumping station the water is delivered by discharge pumps through two discharge pipes to the chemical treatment plant of NPP EBO V2.

Figure 1-2 Overall view of Bohunice site

NPP EMO is located approximately 90 km east from Bratislava. The nearest towns are Tlmače, Levice and Zlaté Moravce. The reference altitude of the power plant ±0.000 m is 242.3 m above sea level. Cooling water for NPP Mochovce is delivered from the Hron river. On the Hron river, there is man-made water reservoir with a total volume of 2.6 million m^3. The reservoir water level is 175 m above sea level at the maximal operating level and 171.5 m above sea level at the minimum operating level. From the reservoir the raw water is supplied to NPP EMO. The water is pumped from the pumping station by approximately 5 km long pipeline into two water tanks of 6,000 m^3 and then gravitationally to NPP EMO.

Figure 1-3 Overall view of Mochovce site

Both sites are connected to the grid by redundant lines. In both cases there are two independent power lines from 400 kV distribution network and two independent lines into the back-up transformers of 110 kV substations. Similarly, in both cases there is a possibility of connecting the power plants to alternative power supply sources from hydro-power stations (different for each site).

Basic data on NPPs are summarized in Table 1-1. All nuclear units in Slovakia have pressurized water reactor WWER 440/V213 manufactured by Škoda company in former Czechoslovakia, with relatively low thermal output of the reactor, 1,471 MWt. The reactor cooling system is located in a large pressure-suppressing containment. Six loops are connected to the reactor, each of which is equipped with main gate valves, and horizontal steam generators with high-volume of coolant on the secondary side of the steam generators. The reactor core consists of 349 hexagonal fuel assemblies; each assembly has 126 fuel rods. 37 control assemblies has fuel parts under neutron absorption portions, so that the efficiency of emergency shutdown of the reactor is increased by ejecting a portion of fuel from the core and at the same time by inserting control assemblies. All units have two steam turbines. Electricity is generated in synchronous generators on a common shaft with turbine and exciter. The power from each reactor unit is fed to the grid through two parallel lines, always from the main generator through the relevant unit transformer with auxiliaries. Both lines branches are connected to output substation to a single 400 kV line.

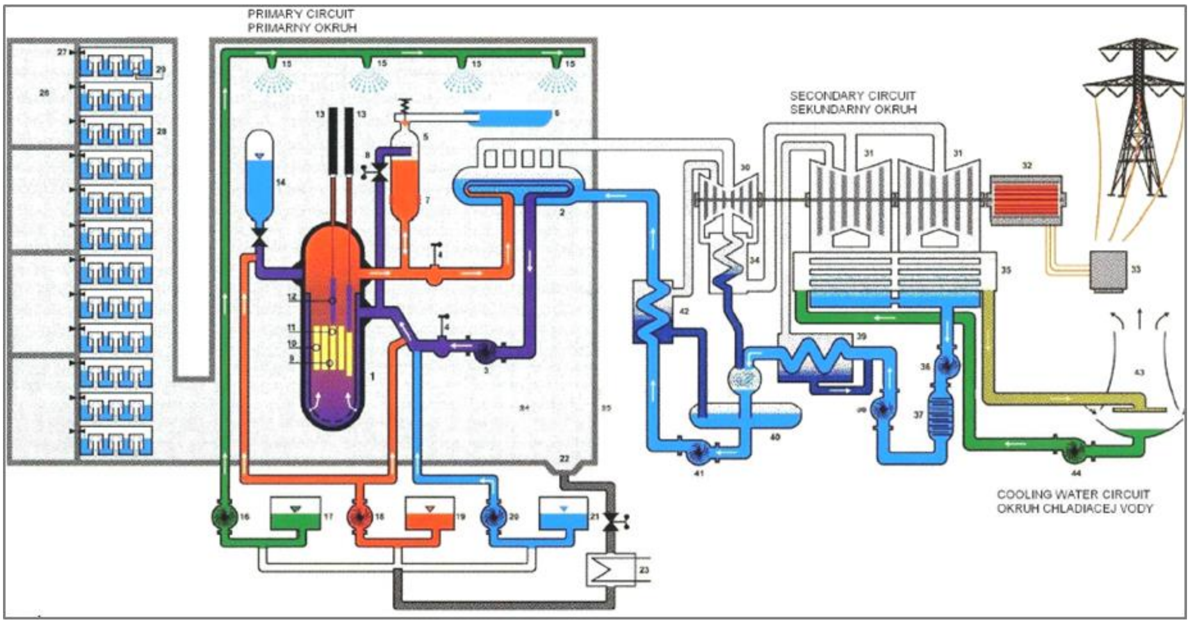

Figure 1-4 Overall scheme of WWER 440/V213

1 – Reactor, 2 – Steam generator, 3 – Main circulation pump, 4 – Main gate valve, 5 – Pressurizer, 6 – Bubbling tank, 7 – Pressurizer, 8 – Pressurizer injections, 9 – Reactor core, 10 – Fuel assembly, 11 – Control assembly (CA), fuel part, 12 – Control assembly (CA), absorption part, 13 – CA drives, 14 – Emergency storage tank, 15 – Spray system, 16 – Spray pump, 17 – Storage tank of the spray system, 18 – Low-pressure emergency pump, 19 – Storage tank of the low-pressure emergency system, 20 – High-pressure emergency pump, 21 – Storage tank for the high -pressure emergency system. 22 – Suction from containment, 23 – Spray system cooler, 25 – Containment, 26 – Air traps of bubble condenser tower, 27 – Check valve, 28 – Bubble condenser tower, 29 – Bubble condenser tower trays, 30 – HP part of steam turbine, 31 – LP part of steam turbine, 32 – Power generator, 33 – Unit transformer, 34 – Separator and steam super-heater, 35 – Condenser, 36 – Condensate pump, 38 – Condensate pump I°, 37 – Unit condensate treatment, 38 – Condensate pump I°, 39 – LP heaters, 40 – Feed water tank, 41 – Main feed water pump, 42 – HP heaters, 43 – Cooling tower CCW, 44 – CCW pumps.

WWER 440 units were designed as dual units with mirror space arrangement. Most systems and equipment belong to one unit; part of systems and equipment is common for both units. Common parts of systems and structures include: fuel-loading machine, shipment of spent fuel, radioactive waste handling, intake and storage of fresh fuel, stack, access to controlled area, demineralized water treatment system, service water system and cooling water system. Each unit is equipped with its spent fuel pool, which is located near the reactor pressure vessel. The spent fuel is cooled in the spent fuel pool (in a compact storage grid in a pool filled with boron water) for about 4 to 7 years.

Table 1-1 Basic data on NPP

Power Plant	NPP EBO V2	NPP EMO	NPP MO3&4
Site	Bohunice	Mochovce	Mochovce
Reactor type	VVER 440/V213	VVER 440/V213	VVER 440/V213
Reactor thermal output, MWt	1471	1471	1375
Total electrical power, MWe	505	470	470
State of the power plant	in operation	in operation	under construction
Date of first criticality	1984-85	1998-99	2018-19 (planned)
Last periodic safety review	2008	2011	-

1.2 Process of preparation of the National Assessment Report

Based on the Council Directive 2014/87/EURATOM and Atomic Act (Act No. 541/2004 Coll.) the licence holder is obliged to carry out a TPR based on UJD SR's specification.

Coordination of NAR preparation in Slovakia was provided by ÚJD SR. ÚJD SR, by an internal order of the Chairperson No. PP 240 002:17 set up a NAR working group at ÚJD SR and invited the licence holder to supply materials for the preparation of NAR. The Chairperson´s order included the intention, the reasons for the work activity, the methodology, the time and content schedule of tasks to prepare the NAR, the boundaries, the resources allocated to preparation of NAR, the powers and responsibilities for the designated staff of ÚJD SR, the way of communication between the members of the working group and with the licence holder, the expected outputs, the milestones, potential risks, etc. The working group was a cross-sectional group of ÚJD SR staff from various organizational units, who have experience in preparation of NAR and are involved in the assessment and inspection activities of ÚJD SR in the area of ageing management. ÚJD SR determined the scope and content of NAR according to WENRA specification [4] and discussed it with the licence holder. Several working meetings of the members of the working group and the licence holder took place during the preparation of the NAR. Meetings were focused on co-ordination of activities, exchanging information on the progress and status of NAR preparation, and addressing open issues. The outcomes of the meetings were documented. The NAR was prepared on the basis of Slovak legislation and documents provided by the licence holder, the contributions from the members of the working group, the results of activities of ÚJD SR, as well as external assessments. The ÚJD SR evaluated and completed submittals to NAR provided by the SE, a. s., added its own assessments, compiled and finalised the NAR. The NAR was approved at the ÚJD SR Chairperson's meeting, then translated into English language, distributed to the involved parties, delivered to the EC and published at the ÚJD SR website.

The preparation of NAR was preceded by the specification of the content and scope of the NAR, which was done by ENSREG supported by WENRA and its working group RHWG. The permanent members of WENRA RHWG are also representatives of ÚJD SR, who also participated in the development of specification.

In the Slovak Republic only Slovenské elektrárne, a. s. is the holder of licence for operation of NPPs. For the organizational arrangements of the NAR development, an IMS document SE/NAR-02/2017 "Preparation of the National Assessment Report on Ageing Management at NPPs in the SR" was issued in SE, a. s. Ageing management is part of the main process "Generation" and from the organizational point of view it is included under Nuclear Engineering Department. The NAR was prepared by internal staff of the nuclear engineering and engineering support departments of SE, a. s.

2 Requirements and implementation of the overall ageing management program

2.1 National regulatory framework

The basic requirements of ÚJD SR for the development, introduction and implementation of SSC ageing management programs are set out in ÚJD SR Decree No. 430/2011 [19] and the safety guide [5]. This guide is based on recommendations of the IAEA requirements [9], safety guide [1] and WENRA requirements [2].

For the ageing management, ÚJD SR requires regular reports according to the Decision [3] from NPP EBO V2 and NPP EMO concerning:

- Remaining service life of the reactor pressure vessel and classified unit equipment, including critical temperature of the brittle fracture of RPV,
- Assessment of the critical brittle temperature of RPV based on tests of chain of surveillance RPV specimens,
- Evaluation of in-service inspection programs.

The requirement to conduct periodical, comprehensive and systematic nuclear safety review (PSR) is given by the law [6], according to which the licence holder is obliged to carry out PSR taking into account the current state of knowledge in the field of assessment, and take measures to remove the deficiencies identified. Areas, for which PSR is performed, are defined in the ÚJD SR Decree [7]. To the reviewed areas belong also the areas of "SSC Ageing Management" and "Operation of the Nuclear Facility after Reaching the Design Service Life".

Requirements for the implementation of work needed to ensure the safe long-term operation (LTO) of nuclear installations are defined in the ÚJD SR safety guide [8].

2.2 International standards

The basic IAEA document dealing with ageing management is a safety guide [1]. The recommendations of this guide are used in the development, implementation and improvement of SSC ageing management programs relevant for the nuclear safety of NPPs in SE, a. s., including recommendations for the key attributes of an effective ageing management program. Effective ageing management over the entire service life of SSCs requires the use of systematic approach to ageing management. The safety guide [1] includes recommendations needed to meet the requirements defined in the IAEA safety standard [9].

For the periodic safety review of NPP EMO and NPP EBO V2, which the licence holder is obliged to perform under the law [6], for the ageing management area, recommendations and instructions given in [10] are used.

When formulating the guiding principles, sequence and the content of individual steps in the LTO program, recommendations [11] are used. These recommendations are part of the safety guide [8] and SE, a. s. methodical guides [12], [13].

2.3 Description of the overall ageing management program

2.3.1 The scope of the overall AMP

2.3.1.1 Responsibilities of specialized departments in the AMP process

The organizational structure of the licence holder for nuclear units is given by the process model of the company, SE, a. s.

To implement the ageing management in SE, a. s., an Ageing Management Group – Nuclear, was set up. Organizational integration of the Ageing Management Group is shown in Annex 1. Ageing management is included in the process model of the licence holder in the top process "Generation". The owner of this process is the Production Division Director**.** In the top process Generation it is included in the Engineering process**,** the owner of which is the Director of Nuclear Engineering. The process is classified as key (main) process. In Engineering, the Ageing Management Group is included under Nuclear Projects Engineering, which co-ordinates and provides all necessary inputs for ageing management activities from other units of the licence holder.

Detailed description of ageing management activities is shown in Annex 2, resulting in the following responsibilities:

- Nuclear Projects Engineering unit is responsible for:
  - Coordination of SSC AM activities
  - Defining SSC scope for AM
  - Defining documentation list for AM
  - Development, management and update of AMP
  - Defining requirements for collecting and archiving data on SSC status
  - Providing for preparation of corrective actions
  - Feedback from fulfilment of AMP
  - Ensuring development of in-service inspection programs
  - Providing results of SSC in-service inspections
- Engineering Support for the power plant is responsible for:
  - Providing data from operational measurements
  - Providing data on SSC repairs
  - Providing data on inspections, tests, surveillance and certifications
  - Providing data from overall SSC assessment
  - Issues reports in the SAP system to replace individual SSCs for inclusion in the maintenance plans
  - Defining and updating technical points in the SAP system
- Operation is responsible for:
  - Providing information on chemistry regimes and analyses
  - Transfer of surveillance RPV specimens for evaluation
  - Loading and unloading chains of surveillance specimens to/from RPV
- Maintenance is responsible for:
  - Providing results of in-service and technical inspections of SSC
  - Providing data on maintenance, repairs and reconstructions of SSCs
- Safety is responsible for:
  - Providing records on operating modes of the nuclear unit
  - Providing overviews on lapse of criteria according to LCO

2.3.1.2 Scope of AMP documentation

The basic document defining ageing management activities of the licence holder for operation of nuclear units is a methodological instruction [14]. This instruction is valid for all nuclear units in Slovakia, i.e. units of NPP EBO V2, NPP EMO in operation and units of NPP MO3&4 under construction. It establishes rules that provides for the requirements and recommendations of ÚJD SR and international standards.

The follow-up documents to this methodological instruction are individual SSC ageing management programs, which take the form of guides. The licence holder has a total of 19 ageing management programs for individual SSCs listed in Annex 3. The programs are developed and binding for units of NPP EBO V2 and NPP EMO in operation. For units in the construction phase, NPP MO3&4, the relevant AMP will become effective as of the date of physical start-up of unit 3 of NPP MO3&4.

When developing ageing management programs for SSC of NPP WWER 440, the experience from the project "Ageing management and optimization of service lifetime of units of NPP WWER 440", are used.

2.3.1.3 Selection of SSC for ageing management

The basic step in the ageing management process is to determine the range of SSCs under monitoring. Selection of SSCs for ageing management is based on the following criteria:

- Fulfilling the safety function of SSC,
- ÚJD SR requirements,
- WENRA requirements,
- Equipment qualification,
- Experience from NPP operation,
- Relationship of SSC to long-term operation,
- Results of research tasks,
- IAEA IGALL international project outputs.

The result of SSC definition for ageing management is the "List of SSCs for AM", which is defined for nuclear units of NPP EBO V2 in the document [15], for NPP EMO in the document [16] and for NPP MO3&4 in the document [17].

The selection of SSCs for AM of NPP EBO V2 and NPP EMO was performed within the LTO program according to the methodology [18]. This methodology is fully in compliance with the requirements of the ÚJD SR legislation and the IAEA guide [11]. In accordance with the requirements of the IAEA guide, in the selection and assessment for the LTO are included:

- Equipment providing important safety functions – this group of SSCs, in accordance with the Slovak legislation[19] on classification of classified equipment into safety classes, is fully covered by the range of equipment safety class BT I to BT III,
- Equipment that help mitigate certain types of events the function of which resulted from safety analyses (e.g. use of buffer on the piping systems in case of seismic events),
- Equipment important for the LTO – based on special request from the licence holder.

In the completion of nuclear units of NPP MO3&4, in all stages of design, the requirements relating to ageing management of SSC were taken into account. This was done already within the revision of the basic design in the document "Principles of Development and Implementation of Ageing

Management Programs for Units of NPP MO3&4“ and also by development of safety concepts for the most frequent degradation mechanisms. These concepts included the specifics of the NPP MO3&4 project and experiences from implementation of AMP on units of NPP EBO V2 and NPP EMO in operation. At the stage of the implementation project, specific projects of those supplies were elaborated that are necessary in implementation of individual AMPs (the RPV surveillance program, monitoring of the thermal ageing of primary circuit materials, corrosion loop for monitoring corrosion processes in primary circuit materials, erosion corrosion monitoring on the components of secondary circuit, surveillance program for monitoring cable life).

Ageing management programs are developed primarily for a category of equipment called “Long-life Equipment“. This is equipment that is not expected to be periodically replaced over the operation of the NPP. These are those pieces of equipment that would be complicated, costly or high procurement cost to be replaced.

Short-life equipment is monitored by the maintenance or qualification programs. Activities under the maintenance program must be scheduled to detect component deterioration before failure [20]. Maintenance activities are described in the methodological instruction [21].

The licence holder has established an “Equipment Reliability” process, in which the NPP is divided into systems; the system being defined for the purposes of equipment reliability as a group of equipment that together provide a defined design function. For each system, a “System Maintenance Strategy” is developed, which presents documented preventive maintenance for a particular system, including a list of system functions, system components, their criticality, and maintenance templates associated with system components.

A description of maintenance activity for individual components is provided in the “Maintenance Templates” (MT), which contain a list of types and descriptions of preventive and predictive maintenance activities (monitoring, tests, inspections, routine repairs and general overhauls, checks, diagnostic checks for faults) that are performed on critical (A) and non-critical (B) components of the same or similar type (e.g. valves, pumps, electric motors) indicating its frequency. The recommendations of the manufacturer and STD are utilized in development of MT. The manufacturer´s requirements, which are stated in the STD in connection with the maintenance of the equipment after modifications within the warranty period, are superior to the approved MT. Requirements for maintenance of components of qualified equipment resulting from quality plans and IQAP are binding, and are superior to the MT requirements and are taken into account when developing and updating MT. Maintenance templates are the basis for development of SMS.

The main parts of the “Maintenance Templates” for the given component are:

- Component definition,
    - Structural parts
    - Usual causes of failures
    - Elements contributing to the risk of maintenance performed
    - Definitions – importance, work cycle, operating conditions
        - Importance – breakdown to critical/non-critical component
        - Definition of the work cycle – high/low
        - Operating conditions – difficult/normal
- Description of maintenance activities – listing the activities to be performed on the component. The following points are described for each activity:
    - The objective of the activity,
    - Content of the activity,

    - Main locations and causes of failures,
    - Development of degradation over time,
    - Support for task interval and relationship to other tasks.
- Table of maintenance activities with the description and interval of maintenance activity, fault codes, sources of required activities (program, law, inspections) for a given work cycle, working conditions, criticality and component category.

Separate maintenance activities prescribed in the maintenance templates are performed in accordance with maintenance procedures in accordance with the guide [22].

Equipment reliability process integrates and coordinates a wide range of activities performed on equipment into a single process [23], which evaluates the status of important facilities.

Inputs for defining the scope of monitoring are set out in the methodological instruction [24]:
- Identified system functions,
- Identified critical, non-critical components and components run to failure,
- SMS documents,
- Maintenance template documents,
- Requirements from the SNaP process,
- Monitoring history,
- Program outputs (AMP, sealing program),
- Feedback from implemented preventive and corrective maintenance,
- Feedback from implemented SUP,
- Data from stable on-line diagnostic systems,
- Data collected in SAP,
- Process of continuous increase of equipment reliability.

The collected and evaluated data are used to develop system and component sub-group “Health Reports“.

A system “Health Report” typically includes:
- Current state of system performance and its trend,
- General justification for system performance evaluation,
- Description and structure of the system,
- Scorecard system,
- Outputs from SUP and AMP,
- Problem analysis (description, risk, strategy, proposed measures),
- Conclusion.

“Health Report” of the component sub-group typically includes:
- Current state of component sub-group performance, assessment and trend,
- Description and structure of component sub-group,
- Basic failure modes,
- Preventive maintenance strategy,
- Reliability analysis,
- Problem analysis (description, risk, strategy, proposed measures),
- Conclusion (assessment of the state of component sub-group with the view to the future).

Based on assessment of individual systems and component sub-groups and outputs from the long-term planning process as defined in the methodological guide [12] the “Report on the State of the Power Plant Equipment” is developed.

2.3.2 Assessment of ageing

Individual ageing management programs have the following structure according to the methodological instruction [14]:

- Description of basic information on SSC,
- Identification of degradation mechanisms,
- Collection and recording of data,
- Assessment of the current status of SSC,
- Preventive and corrective actions.

2.3.2.1 Description of basic information on SSC

Each individual AMP for SSC contains the following basic data on SSC:

- Description of the structure and function of SSC,
    - Basic information on SSC – name, type, location, technological identification, safety function
    - Technical parameters of SSC – manufacturer, performance, description of structure, technical conditions, serial number
    - Description of the function and use – description of the SSC function and basic requirements for SSC functionality, safety and qualification
- Description of material characteristics,
    - For each SSC given type of material, material composition
- Description of SSC operating conditions,
    - The parameters of the operating environment (pressure, temperature), ambient parameters (temperature, pressure and humidity) during normal and emergency conditions.

2.3.2.2 Identification of degradation mechanisms

The method for identifying degradation mechanisms affecting SSC is described in the methodological instruction [14] in chapter “Identification of degradation mechanisms“. It is performed on the basis of knowledge of:

- Basic design,
- Generation,
- Mode of operation,
- Operating diagnostics,
- Maintenance program,
- Program of in-service inspections,
- Experience of other operators,
- Latest knowledge of science and research.

In individual SSC ageing management programs, the following information is included in the description of degradation mechanisms:

- Description of degradation mechanism and identification of degradation locations
    - A list of degradation mechanisms affecting the SSC
    - Definition and description of the critical degradation mechanism (or several mechanisms, if applicable)

    - Determining the method of selecting locations with the most effects of the degradation mechanism.
- Determining the methodology for assessment of the degradation mechanism
    - For degradation mechanisms there are assessment methods defined [25] including the definition of status indicators and determination of the acceptability criteria.

Identification of the degradation mechanisms for the purposes of LTO of NPP EBO V2 was realized on the basis of the documents [18] and [25].

The acceptability criteria for individual degradation mechanisms are set out in the individual AMP. They are designed in accordance with the methodological instruction [14] to ensure safe and reliable operation of SSCs.

2.3.2.3 Data collection and recording

For the purposes of assessing the current state of SSC, the collection and recording of data necessary for the quantification of effects of degradation mechanism, is performed. The input data are provided by specialized departments from the following sources (see Annex 2):

Test Database (SUP) – Support application for the mode management group. It is used to record and archive the results of the surveillance tests.

DATD – Digital archive of STaPD unit. It is used to record, archive and provide documentation on the construction. Access is secured through DATD-NET application.

PREV-DOK – Network application of STaPD unit. It is used to record, archive and provide operational documentation.

DEFK (NPP EBO V2), ASSIK (NPP EMO) – Applications of the NDT unit. They are used to record, archive, evaluate and provide records (protocols) from performed inspections.

CHEMIS32 – Database application of specialized department of chemical inspection. It is used to process data from laboratory assessment of samples and continuous inspection measurements.

TPS (NPP EBO V2), BIS (NPP EMO) – Multilevel information and control system. It serves for direct measurements and processing technological data from the secondary, primary and tertiary circuits of the NPP and for communication with other information and control systems. The processed data are archived and presented to the operators in the control rooms and other staff of NPP.

SAP-QM – Quality management application consisting of several modules. The PM Module is designed to archive and evaluate the protocols from quality checks and maintenance interventions.

Program for recording and evaluation of reports from inspections and tests of qualified technical equipment and classified equipment. This database application is used to record, archive and evaluate protocols from the performed activities.

Source databases are regularly updated by the relevant source database administrators. The intervals for collecting and recording data needed to assess the SSC status are defined in individual AMP.

For the ageing management there is a database application "Ageing Management Databank", which is used to record, archive and evaluate SSC data, needed to assess the current state of SSC included in each AMP. Database management is performed according to special regulation [26] and corresponds to the rules for documentation management set out in the methodological instructions [27], [28]. The database is maintained by the staff of the "Lifecycle Management Group", including managing access to the database.

The originals of the protocols from the inspection activities in the area of in-service inspections, technical inspections, inspections and tests on qualified technical equipment and diagnostics, are sent by the responsible units to the technical documentation registry centre at NPP EMO and NPP EBO V2, where they are stored throughout the entire life of the power plant.

All records are kept in accordance with the methodological instruction [29].

The records are a special group of documentation that is produced during the operation of NPP EBO V2 and NPP EMO. They may take a form of completed forms, operational logbooks, protocols, registration slips, digital recorders, etc.

Record management is performed according to the rules set out in the methodological instructions [27], [28].

In cases where records are not centrally managed, but their management is within the competence of specialized units, the record management process is part of the documents, under which activities are performed (instructions, operating instructions, work procedures, special procedures).

Records of operational management documentation (records from meetings, decisions) are managed from the level of the company management (SE, a. s.) according to the methodological instruction [29].

2.3.2.4 Assessment of the current state of SSC

For the assessment of the current state of SSC, there is:

- Methodology for degradation mechanism assessment,
- Description of measured values to determine the degree of SSC degradation,
- Setting the interval for degradation mechanism monitoring.

The assessment of the degradation mechanism according to the relevant assessment methodology is provided by the Lifecycle Management Group. In case that the assessments, calculations and laboratory measurements cannot be performed by the Lifecycle Management Group, these are processed by an external organization. SSC assessments, including assessment criteria and intervals, are described in individual AMPs. The results from internal and external analyses and calculations are summarized by the Lifecycle Management Group.

By comparing the parameters of the current status of SSC with the status in the previous periods under assessment, trending of material degradation in the monitored SSCs is performed.

As part of the equipment reliability process, SE, a. s. has a system of assessment and trending of performance (Annex 4).

In case of change in the parameter, the following measures are implemented:

- If the parameter status is improved, it is analysed whether this change is stochastic (random) or it is a response to a corrective action. The facts, based on which the parameter values have changed, are briefly described in the Scorecard or the Health Report.
- In case the parameter deteriorates, an analysis is performed. If incorrect monitoring (incorrect setting of parameters or criteria for parameter monitoring) is not the cause of the deterioration in the parameter status, the reasons for deterioration are identified, the reasons of parameter change are found and the methodological instruction is followed [30]. The result of the analysis is stated together with a short comment, in the Health Report.
- If the overall system/ component sub-group status is improved by two colour scales, the colour evaluation and the trend of improvement of the overall status of the system/component sub-group is made in the Health Report. This fact is also explained with a comment.
- If the overall status of the system/ component sub-group deteriorates, an analysis is performed and an action plan is adopted to make improvements.

If the status of SSC is unchanged, the trends of individual parameters are checked. If the system/component status is acceptable currently, it is checked whether the trends are not showing deterioration.

The results of assessments are discussed by the Equipment Reliability Commission and provided to the NPP management.

In accordance with the methodological instruction [14] for the classified SSC included in the AMP, the following evaluation documents are produced at specified intervals:
- Annual assessment report on the status of SSCs – the basis for this report is results and activities from individual AMP of classified equipment. The report is drawn up in the following scope:
  - Evaluation of the current status of SSC included in the AMP based on the specified status indicators
  - Evaluation of meeting the time schedules for assessing individual AMP for SSC
  - Draft measures to eliminate or mitigate the effects of degradation mechanisms on SSC
  - Assessment of effectiveness of measures taken
  - Proposal for planned AMP activities
- Report on periodic safety review for the area of ageing management – the report is prepared in accordance with the current legislative requirements [6] and [7]. Evaluated aspects for the ageing management area are:
  - Ageing management programs strategy and documentation
  - Completeness of the list of classified equipment included in the AMP
  - Records and the suitability of selecting recorded data affecting ageing, as well as data identifying the status of lifecycle of classified equipment
  - Results of lifecycle monitoring and effectiveness of AMP of replaceable classified equipment
  - Acceptability criteria, current and required safety margin of classified equipment
  - Level of understanding the physical conditions, prevailing ageing mechanisms, current safety margin and other effects that could reduce the lifecycle of classified equipment
  - Possibilities for mitigating consequences of ageing process of classified equipment
- Upon completion of the general outage of each unit “Report on Lifetime use of Selected Unit Components” is submitted to ÚJD SR. In case of a more pronounced adverse change in the value compared to the previous year, the Report states the causes and proposed, prepared or already implemented corrective actions.
- Before the physical start-up of the unit after the GO, there is a meeting of the Commission for evaluation of QP results of classified equipment, presenting information on the following:
  - Course of unit GO, realized design changes

- Results of the in-service inspection program (NDT and technical checks)
- Results of inspections and tests of technical equipment
- Results of inspections and work on containment
- Results of diagnostic systems
- Results of lifetime use (RPV, cables, and ESW pipes).

Presented results are processed in the report “Evaluation of Results of the Quality Program of Classified Equipment of the Unit Performed during GO” and are sent according to the relevant Decision [3] to ÚJD SR.

2.3.2.5 Use of research and development programs

The licence holder participated as the implementer for the R&D task 1300 “Ageing Management and Optimizing the Lifecycle of NPP Units with WWER 440“. The project results are used to create individual AMP and a systematic approach to ageing management.

All innovation activities of the licence holder are coordinated in accordance with the methodological instruction [31] by the Innovation Centre, which is part of the Engineering unit. This Centre has a coordinating role and also plays a role of a single point of contact for topics and activities in the field of innovation that take place at the level of operated nuclear units of NPP EBO V2, NPP EMO and NPP MO3&4.

At present, an intensive exchange of information on new knowledge and research results in the area of SSC degradation is under way within various projects of research organizations and NPP operators. The licence holder is permanently represented in these projects, ensuring the transfer of information to the AM processes at the NPPs.

The licence holder is represented in the following projects:

- In the period 2005-2007 the contractor involved in the activities in the field of SSC ageing management, participated in the SALTO project aimed at developing basic LTO principles, mainly on ageing management processes. In 2012, representative of the licence holder took part in the SALTO mission in Paks NPP as an observer for the area “Building Structures“.
- Since 2010, the licence holder has been participating in the IAEA IGALL project – Project is aimed at developing a practical manual for ageing management of NPP equipment important in terms of nuclear safety, including recommendations for effective management of ageing programs. The Project is divided into three professional fields: machinery, electric equipment and building structures of NPPs. The licence holder is represented in each working group.
- OECD/NEA CADAK – Project is aimed at expanding the existing databases and knowledge in the field of ageing management and cable qualification. The licence holder has been involved in this project since 2011.
- OECD/NEA CODAP – Project aimed at expanding the existing databases with failures of passive components of the primary circuit, as well as other components whose failure has a significant effect on the operation, including measures from the viewpoint of degradation mechanisms. The licence holder has secured access to the database of operational experience, as well as the possibility to contribute with information on events occurring in the operated NPPs.
- VERLIFE IAEA – A part of this project was also development of the Manual for Assessment of Integrity and Life of Components and Pipes in WWER type NPPs [32]. After validation in individual countries operating WWER NPPs, the Manual will be used for assessment of integrity and life of selected SSCs. The application of this Manual will ensure unification of the calculation

procedures and the assessment methodologies for possible comparison of results and additions to the knowledge database on the degradation of structural materials of WWER NPPs.

- The IAEA – Research Coordination Meeting on the Review and Benchmark of Calculation Methods on Piping Wall Thinning due to Erosion – Corrosion in Nuclear Power Plants. The Project lasted from 2011 until 2014. The output from the Project will be a manual for erosion corrosion monitoring, including theoretical aspects of erosion.
- OECD Halden Reactor Project – Fuels and Materials – Project is focused on the development of Small Punch Test methodology, which allows the determination of basic mechanical properties using small samples. The aim of the projects was to directly correlate results with the current surveillance specimen programs for NPP EBO V2 and NPP EMO, and to assess the impact of irradiation of samples in the energy and research reactor on the rate of degradation of mechanical properties.
- Supplier organization involved in the activities in the field of ageing management and cable qualification was one of the parties in the IAEA project “Coordinated Research Program on Qualification, Condition Monitoring and Management of Ageing of Low Voltage Cables in Nuclear Power Plant Life Management“, 2012-2015. The main aim of the project was to identify monitoring methods with potential for monitoring of degradation and assessment of ageing of different types of insulating materials.
- The licence holder has access to the EPRI (Electric Power Research Institute) databases and materials in the field of ageing management.
- The licence holder is a member of the International Equipment Reliability Working Group focusing on exchange of experience in the Equipment Reliability process.

2.3.2.6 Use of internal and external experience

The licence holder has a detailed procedure for obtaining, analysing and applying operating experience described in the documents [33], [34], distinguishing between external and internal operational experience. When using operational experience the principle applies that external operational experience is taken as an internal experience.

The source of experience is usually:

- Internal experience of the NPP
  - Database of internal events, events without consequences (low level events) and near-misses (SAP)
  - Corrective actions database (SAP), JIT (Just in Time) database – summary of information on operational experience (SAP) database of operational logs.
- External experience
  - Database of experience, reports from WANO/INPO/IRS/ČEZ/SE (SAP)
  - Database of preventive measures (SAP)
  - Other relevant WANO/INPO documents
  - Information from communication with other NPPs and information from suppliers
  - Cooperation with the coordinator for use of operational experience.
- Other sources of experience
  - Electronic dissemination for immediate notification of selected groups of staff, periodicals, magazines, information centres (notice boards), video information system of NPP.

In addition to the above mentioned use of operational experience, the licence holder also applies the principles of self-assessment and benchmarking in accordance with the directive [35]. The self-assessment and benchmarking program is based on the basic principles set out in the WANO guide [36].

Benchmarking in the operation of NPPs is used for systematic comparison of procedures, methods and management expectations with other high-performance organizations to identify best practices and methods and to set up a performance or process improvement plan.

Self-assessment is the assessment of specific activities, facilities, units, performance areas, programs, processes or systems against specific criteria. Self-assessments are used to compare the current performance against the highest standards of generation, management expectations and requirements of regulatory bodies, in order to identify and to remedy areas requiring improvement.

The licence holder has an established process of “Use of Operational Experience“. Outputs from this process are used in the “Ageing Management” process. On the basis of important knowledge from the operation, equipment maintenance and checks, suggestions are developed to eliminate excessive damage to individual SSCs backward in the AM process.

In addition to the above sources from internal and external experience, the following events are regularly organized:

- “Seminar to Exchange Experience in the Field of Ageing Management and the LTO“ with the participation of specialists from SE, a. s., ČEZ, a. s., ÚJV Řež, a. s. and VÚJE Trnava, a. s.,
- Reliability Management Meetings – Slovakia, Czech Republic, Hungary,
- Benchmarking “Database of Chemical Regimes“ – Slovakia, Czech Republic, Hungary, Finland,
- Peer reviews and WANO technical support missions,
- OSART reviews and the IAEA technical support meetings.

2.3.3 Monitoring, testing, sampling and inspection activities

These activities are covered by the licence holder within the ageing management process are as follows:

- Surveillance programs to monitor the status of SSCs
  - RPV surveillance specimen program
  - Program of monitoring thermal ageing of primary circuit equipment
  - Program for monitoring ambient parameters for implementation of cable AMP
  - Cable surveillance specimen program
  - Program to monitor cable fire-proofing coating
  - Program to monitor the corrosion status of primary circuit materials – Corrosion loop
  - Program to monitor corrosion status of secondary circuit materials.
- Program of in-service inspections [37], [38], [39] has been introduced by the licence holder from the start of operation of nuclear power plants. The scope and frequency of testing, monitoring and inspections of classified equipment are determined by their safety relevance and are based on valid quality plans of classified equipment [40], unit Limits & Conditions, the national legislation [6], the IAEA recommendations, operational experience, long-term operation requirements, recommendations resulting from the results of computational analysis of the damage, the introduction of the new qualified methods of testing and changes in the SSC inspection intervals. Other sources for the elaboration of in-service inspection programs are technical conditions or equipment operating instructions. On the basis of the above documents, a workflow [41], [42], [43], [44] is elaborated for each NPP unit. Any changes in the program of in-service inspections, their frequency, the method of assessment, are approved by ÚJD SR.
- Programs of pre-operational inspections of SSC at NPP MO3&4 units under construction include operational experience from NPP EBO V2 and NPP EMO units in operation. Within pre-operational inspections of SSC of NPP MO3&4, not only the results of NDT inspections already

carried out were collected, but also zero measurements of all SG heat-exchange tubes, secondary circuit piping components included in the monitoring within AMP for piping of secondary circuit, samples were taken from the secondary circuit components to verify the chemical composition, prepared surveillance cable specimens for laying in real operating conditions. Data from these measurements are registered in DRS and will be used for the assessment of SSC lifecycle at NPP MO3&4.

- Monitoring of NPP operation is ensured by the following activities
  - Surveillance Program (SUP) – A program that ensures execution of inspection activities to verify that the nuclear power plant is operated within the prescribed LCO (or other technical specifications), and ensures timely detection of any deterioration in building structures, systems and equipment, as well as any adverse trends that could lead to reducing the level of nuclear safety
  - Functional tests – Scheduled and periodical tests that must guarantee that the system or equipment being tested is capable of performing its designed function, in accordance with the requirements of technological and operational procedures, LCO, regulatory bodies and standards
  - Monitoring and analysing data from the technological process and operational conditions of equipment and proposing corrective actions
  - Analysis, assessment and optimization of modes of normal operation and failure operating conditions of units
  - Testing protection and blocking devices – To verify correct action and setting of levels of automatic devices, protection and blocking devices
  - Operability tests – Trial operation of the equipment to verify their readiness for operation and capability to meet its design functions
  - Development, review and optimization of operational documentation for normal operation (operating regulations, programs, work procedures) to ensure management of unit modes, compliance with the nuclear safety and LCO requirements.

2.3.4 Preventive and corrective actions

The licence holder has a Prevention and Corrective System, which is based on the basic principles of WANO guideline [36]. The basic requirements and rules are set out in the methodological instruction [45]. The aim of the SNaP process is to prevent severe operational incidents through identification of issues and early removal of their causes and to ensure safety, reliability and economic efficiency of nuclear power plants in operation.

Analysis of problems/ incidents is carried out based on methodologies contained in the methodological instruction [46]. The result of the analysis of causes of the problems is the proposed corrective actions, the implementation of which is ensured in accordance with the methodological instruction [47]. Detailed requirements and procedures for detecting and analysing trends are set out in the methodological instruction [48].

In the case of serious operational events, immediate investigation of causes of event together with corrective actions taken to restore the normal unit operation is provided by the "Extraordinary Event Committee". Detailed requirements for its activity are set out in the methodological instruction [49].

Regular monitoring of SNaP performance and effectiveness is implemented through reports that are submitted to the SNaP Commission meeting. On the basis of reports the SNaP Commission takes corrective actions to prevent recurrence of serious problems and to improve the efficiency of the process [48]. The basic scheme of the SNaP process is shown in Annex 5. Corrective and preventive actions that are inputs for the Equipment Reliability process are part of the Health Reports of the individual systems. Administrative checks shall be carried out in accordance with the procedures laid down in the managing documents [50].

Assessment of the suitability of preventive and corrective actions before unit start-up after a regular GO is under the responsibility of the Commission for evaluation of results of in-service inspections, whose activity is described in the guide [51].

If necessary, corrective actions are addressed in the ageing management process according to the guide [14]. For this purpose working groups are being set up to analyse the causes of the increased effect of the degradation mechanism and to propose corrective actions to reduce the trend of the degradation mechanism effect.

In case that an inspection, technical inspection and testing, maintenance or information on findings and failures demonstrates SSC damage due to the effect of another degradation mechanism that has not been considered or a deviation from the design is demonstrated, which was not detected by the use of monitoring means, in accordance with the methodological instruction [14] analyses are carried out, the results of which are used to update the AMP, maintenance or on the basis of which extended repair and maintenance of SSC or a change in operating conditions is initiated.

Assessments, calculations, tests and laboratory measurements that cannot be performed in-house, are processed by an external organization. The results of these analyses and the resulting measures are recorded in the ageing management database.

For the purpose of defining procedures for collecting and processing information about events, investigating the causes of event occurring, taking corrective actions and monitoring their fulfilment during construction, inactive tests and commissioning, the licence holder of NPP MO3&4 under construction has a methodological instruction [52]. For the purposes of feedback from operation, the licence holder for NPP MO3&4 under construction has a methodological instruction [53].

2.4 Revision and update of ageing management program

Updating AMP is performed by the Lifecycle Management Group in a form of revisions in the following cases:

- Change in the methodology or criteria for the assessment of degradation mechanism,
- Identification of new SSC degradation locations,
- Experience from AMP implementation,
- Operational experience,
- Audit findings,
- Design changes,
- Change in the organizational structure,
- Time limited analyses,
- New qualified test methods,
- Periodic safety review.

The licence holder reviews and updates on a regular basis, at least once every three years, the individual AMP in accordance with IMS SE, a. s. and the ÚJD SR Decree [19]. If this review results in a need to update the documentation that is not under the ÚJD SR´s review or approval, the licence holder shall make its revision within three months from such review.

Ageing management process is evaluated every 10 years in a form of "Ageing Management Assessment Report", which is part of the PSR assessment report in accordance with the ÚJD SR Decree [7]. Upon PSR completion the licence holder implements the action plan for corrective actions.

The assessment and update of AMP was also carried out within the framework of the Long-term Operation Program of NPP EBO V2 in accordance with the document [54] in the scope of legislative requirements [7]. The purpose of the review was to identify suitability, sufficiency and completeness of each established AMP.

Outcomes from the assessment are the AMP validation protocols. In accordance with the conclusions and recommendations from the assessment protocols and sheets, an "Action Plan of Corrective Actions of the LTO program of NPP EBO V2" is developed.

2.4.1 Evaluation of power plant modifications that may affect the overall AMP

The licence holder has an established "Configuration Management" process with the following sub-processes:

- Configuration management of the initial process – "Design Authority" (DA),
- Management of technical changes in NPPs,
- Maintaining of SAR,
- SNaP management,
- Managing the process of operation and maintenance of NPP technology.

"Design Authority" is focused on independent supervision in the process of technical changes. It has the obligation to check, verify and approve or reject technical changes to SSCs.

The technical change is initiated through the SAP Nuclear system through the transaction "Entering Proposal for Technical Modification". The solution to initiating the change is ensured by the preparation and subsequent approval of the proposal for change in accordance with the guide [55]. Each proposal for change after its registration and approval by the licence holder´s respective unit is subsequently discussed and approved by the Technical Commission of the operated NPP EBO V2 and NPP EMO units.

In order to transfer operational experience to the nuclear unit of NPP MO3&4 under construction through the Technical Commission of the operated unit of NPP EMO, the proposals for technical change together with recommendation for their implementation during construction, commissioning or operation, are sent to the Configuration of the Basic Design of nuclear unit under construction, NPP MO3&4.

After arrangement of all activities related to implementation of changes from operating costs, including functional tests, the changes are implemented on SSCs, incorporated in the as-built documentation in accordance with the methodological instruction [56]. Based on the changes made to SSCs, individual AMPs are updated if necessary.

All design changes classified as "Significant Changes" and "Minor Changes" implemented during the design, manufacturing construction, installation and commissioning of nuclear units under construction, NPP MO3&4, are recorded in the NPP Design Change Database. Access to the Design Change Database is secured through the Intranet SE, a. s. Other design changes that may occur during the design phase are managed as revisions of individual documents subject to approval by the Engineering unit. The process of modifications to the Basic Design of nuclear units under construction, NPP MO3&4, is stipulated by the Director´s Order [57], describing the system of design change management and its principles.

As part of the SNaP process, the Design Authority process focuses on assessing and preparation of opinions on internal and external operational events and corrective actions in terms of NPP Design Basis [58].

2.4.2 Evaluation and measurement of effectiveness of ageing management

Effectiveness of ageing management of individual SSCs is evaluated based on the assessment of defined status indicators and defined acceptability criteria that are part of all individual AMPs. Information on the identifiers that denote on the status of SSCs under monitoring within the AM, are part of the “Annual assessment report on the status of SSCs“.

In addition to the activities resulting from the AM process the licence holder, in the “Equipment Reliability” process, has established the process “Performance Monitoring“.

For each important function of the system the monitoring parameters are defined that are necessary to effectively monitor adverse trends and degradation that can affect the system performance. The reports on the system status, the Health Report, periodically evaluate the current state of system performance and its trend in accordance with the methodological instruction [24]. System of Performance monitoring and trending are described in par. 2.3.2.4 and chapter 2.4.2.

2.4.3 Evaluation of time-limited analyses

The requirement for a revision of safety Time-Limited Ageing Analyses (TLAA) is defined in the Slovak legislation [7]. TLAA are types of safety analyses specific to a particular NPP that are based on explicitly considered operating time period or design technical lifecycle of the equipment, and include the effects of ageing on the equipment. Revision of the TLAA analyses is carried out for part of SSCs at the NPP identified as “safety relevant long-life equipment”. The purpose of the revision is to identify the relevant TLAAs for each assessed SSC, and to evaluate their validity for the entire estimated LTO period. As with the AMP revision, an assessment methodology [59] is developed and approved before starting with the assessment itself. The record of the course and the results of the review are in the form of evaluation protocols.

2.4.4 The process of incorporating legislative changes to AMP

The licence holder participates in drafting legislation in accordance with [60] and [61]. Management of communication with the relevant regulatory bodies aimed at incorporation of the comments and suggestions of the licence holder in the legislation for nuclear installations shall be conducted in accordance with the methodological instruction [62].

Changes in international legislation are incorporated into the AMP in a form of program revisions.

For the nuclear unit under construction, NPP MO3&4, there is a system of periodical update of information and knowledge on Slovak and EU legislation already approved or under preparation, to allow assessment of their impact on the licensing process of SE-MO3&4 project [63].

2.5 Experience of the licence holder with the application of overall AMP

Ageing management and lifecycle assessment of NPPs started to be implemented in SR from 1991, while it was part of several projects aimed at increasing nuclear safety and reliability of NPP operation [64]. The rules of systematic approach to SSC ageing management were defined in 2001, in the ÚJD SR Guide, the "SSC Ageing Management" [65]. Gradually, individual SSC ageing management programs were developed and introduced.

Individual ageing management programs were developed at the start of the ageing management process of the licence holder. In connection with the long-term operation program, the SSC ageing management programs are developed for a period of 60 years of operation, which the WANO Peer Review mission at the NPP EMO in operation conducted in 2013, and OSART mission (extended with the LTO module) on NPP EBO V2 in operation conducted in 2010, classified as "Good Practice".

As part of power uprate of the nuclear units of NPP EBO V2 and NPP EMO, analyses of impact of the power uprate on the SSC performance were conducted. Individual AMPs were updated based on the outputs from individual analyses. An analysis of the effect of increased power on the secondary circuit pipes was also classified as "Good Practice" by the WANO Peer Review mission at the NPP EMO in operation conducted in 2013.

By introducing the "Equipment Reliability" process, changes have been made in the SSC ageing management process, defining new responsibilities and links to individual departments.

The experience of the licence holder with the ageing management process on the operated units of NPP EBO V2 and NPP EMO was implemented on a nuclear unit under construction, NPP MO3&4. Structural modifications and improvements have been implemented:

- Repair of the upper weld of thermal shielding on CA nozzles of RPV head,
- Replacing rod handles for a new design that allows to increase the allowable hardness value during operation,
- Changing the material of the SEPS SG piping for stainless steel that led to the removal of the heterogeneous joint on the SEPS nozzle of the SG sleeve,
- Change in the SG feed water piping design, thereby moving the heterogeneous joint above the feed water level,
- Changing the material of heat-exchange pipes of condensers from brass to titanium, allowing operation of the secondary circuit with higher pH value,
- Change of material of the critical piping of the secondary circuit for low-alloy steel with higher resistance against erosion corrosion,
- Use of new type of fire non-spreading cables,
- Due to the increased lifetime use of feed water nozzles of the steam generators of NPP EBO V2, an analysis was conducted to find the causes of increased lifetime use. The output of the analysis was the adjustment of operating modes at the stage of unit start-up, which resulted in reduced lifetime use of feed water nozzles of steam generators.

2.6 Regulatory oversight process

The measure to increase nuclear safety of operated reactors of NPP EBO V1 (currently under decommissioning) based on the conclusions of expert opinion on the comprehensive state of this nuclear power plant and required by the former state regulator, ČSKAE, by its Decision No. 5/1991, were the first steps towards activities related to the monitoring of some components of the nuclear installation (NI) in terms of their service life. Based on the results of the analyses, the licence holder has introduced diagnostic systems for the reactor pressure vessel (RPV), the steam generators (SG), the pressurizer and the primary circuit piping. These measures can be considered as implicit ageing management of selected components of NIs. The monitoring methods of these components were used in Slovakia also on operated units of NPP EBO V2.

ÚJD SR, as a newly established state regulator, later defined its requirements and the scope of classified equipment, for which the assessment of the lifetime use needs to be submitted by the licence holder, the scope and periodicity of reports on the lifetime use criteria for fuel for the given campaign and use of limited number of operating modes of the main components of PC, steam pipes and feed water for the given campaign and cumulatively from the start of operation. These requirements were applied on all operating units in the Slovak Republic (NPP EBO V2 and NPP EMO).

The requirements for the process of managed ageing have been systematically emerging among the ÚJD SR requirements since 2001 by issuing an ÚJD SR safety guide, "Ageing Management of Nuclear Power Plants, Requirements, BNS I.9.2/2001, ÚJD SR, Bratislava, 2000".

The current basic requirements of ÚJD SR for the development, introduction and implementation of SSC ageing management programs relevant for safety, are set out in the Safety Guide [5]. This guide is based on the recommendations of the IAEA safety guide [1] and WENRA [2].

Relevant requirements in the Slovak legislation for the ageing management are those stipulated in the ÚJD SR Decree No. 431/2011 Coll., Annex 5 section I, par. g) – Requirements for ageing management of classified equipment in the quality plans for classified equipment. According to this provision the issue of ÚJD SR approval for the operation of NI is conditional on the existence of an ageing management system at NI. ÚJD SR approves the documentation, which includes the quality plan, resp. requirements for ageing management of the selected device, for each change of the documentation during operation of NI, respectively for each new selected device.

The requirement to conduct periodical, comprehensive and systematic nuclear safety review (PSR) is given by the law [6], according to which the licence holder is obliged to carry out PSR taking into account the current state of knowledge in the field of assessment and to take action to remedy the deficiencies identified. The areas, for which PSR is performed, are defined in the ÚJD SR Decree [7]. Areas under review include also: "SSC Ageing Management" and "Operation of Nuclear Installation after Reaching the Designed Lifetime".

In order to harmonize the rules for the preparation and evaluation of outputs from PSR, ÚJD SR has also issued relevant guide BNS I.7.4/2016 Comprehensive periodic review [66], which specifies and complements the requirements of the ÚJD SR set out in the Decree [7]. The basis for development of the BNS were the relevant documents of the International Atomic Energy Agency (IAEA, Periodic Safety Review for Nuclear Power Plants, SSG-25) as well as the reference levels of the Western European Nuclear Regulators' Association (WENRA).

2.7 Regulator´s assessment of the overall ageing management program and conclusions

ÚJD SR establishes the requirements for ageing management programs, assesses ageing management programs of the licence holder and inspects their implementation at NIs in terms of the relevant requirements of the generally binding legal regulations, ÚJD SR Decisions, requirements/recommendations of ÚJD SR safety guides, the IAEA and WENRA safety standards. Assessments and inspections are carried out by the ÚJD SR staff, but also the IAEA and contracted organizations. Assessments and inspections of the ageing management programs are carried out within the licensing process of NI, reviews of compliance with the quality requirements for classified equipment, periodic safety review and other activities. Results of assessments and inspections are documented in the decisions issued by the ÚJD SR, assessment reports, records/protocols from inspections, standpoints and opinions. As a part of inspections, ÚJD SR receives from the licence holders regular reports on the results of program of in-service inspections, lifetime use of classified equipment, classified equipment failures, safety indicators of operation of NI, and others.

Examples of several assessment activities carried out by the licence holder and subsequently also by the ÚJD SR, recently concerning ageing management and overall ageing management program, are summarized in the following text.

Prior to putting the NPP unit into operation after its outage, the results of the program of in-service inspections, as well as assessment of lifetime use of selected components of NI (what is one of regular reports sent to the ÚJD SR) are reviewed by a specific commission. ÚJD SR issues its opinion on the conclusions of this review.

Due to the planned long-term operation of NPP EBO V2, the results of the special periodic assessment of NPP EBO V2 were submitted to ÚJD SR in January 2014 based on the requirement of the ÚJD SR Decree No. 33/2012 Coll. on periodical, comprehensive and systematic nuclear safety review of nuclear installations. ÚJD SR carried out inspection No. 235/2014. The inspection was aimed at reviewing NPP EBO V2 long-term operation documentation (LTO), assessment of existing ageing management programs for individual selected objects, their adequacy in terms of identifying ageing patterns, and how to take corrective action to identify signs of ageing. In co-operation with external experts, ÚJD SR assessed:

- The methodology for selecting systems, structures and components (SSC) for which an ageing management program (AMP) is required,
- AMP review methodology,
- AMP rating results.

The inspection confirmed that NPP EBO V2 has successfully implemented a comprehensive long-term operation program. On the basis of the comments of ÚJD SR on the elaborated PSR, a document entitled "Action Plan of Corrective Measures of the Long-term Operation of the NPP EBO V2" was elaborated by the operator, which contains a set of integrated measures resulting from the AMP-machine part, AMP-electrical part, AMP-building part and AMP-hardware. ÚJD SR acknowledged the Action Plan of NPP EBO V2 for its long-term operation. The fulfilment of the measures was verified by inspections (inspection of ÚJD SR No. 216/2016 and No. 215/2017).

Following the last NPP EBO V2 periodic safety review (PSR), ÚJD SR carried out inspection No. 206/2017, in order to verify the PSR results (including the results of the area – Ageing management).

In the second half of 2017, ÚJD SR carried out an inspection of ÚJD SR No. 905/2017 with the participation of external experts. The inspection was focused on check of ageing management programs in NPP EBO V2 and NPP EMO. The inspection covered following areas:

- AMP of selected building structures,
- AMP of selected electrical equipment,

- AMP of selected machine technology equipment.

During the AMP inspections in the given areas, attention was given to the completeness of the identification of possible degradation mechanisms, status indicators of building structures and facilities and the determination of acceptability criteria, models of prediction of identified manifestations of ageing and corrective measures taken to ensure the operability of construction structures and facilities. In the framework of the inspection, it was inspected:

- State of the data of database on indicators that characterize the various degradation mechanisms (ageing) of construction structures and equipment,
- The completeness of the implementation of the corrective actions of the Action Plan of Corrective Measures of the Long-Term Operation Program of the NPP EBO V2 in the areas of ageing management of building structures, electrical equipment and selected machinery,
- The accuracy and completeness of the summary reports from the PSR NPP EBO V2 (Ageing management and Operation of nuclear facility after reaching its design lifetime).

The assessments conducted by ÚJD SR show that the licence holder has established long-term strategic objectives in the field of ageing management (AM) foreseeing a long-term operation. The AM strategy focuses mainly on activities related to the fulfilment of legislative requirements and ageing management processes, on the development and implementation of long-term SSC rehabilitation programs, and the issue of SSC obsolescence. The documentation also shows that the main priorities and objectives of NPP EBO V2 currently include the implementation of measures for the long-term operation of the power plant, and the continuous improvement of the equipment status. For the implementation of ageing management programs, the licence holder has established a suitable and functioning organizational structure, as well as technical and human resource support to perform all the necessary activities.

The basic legislative requirements are reflected in the IMS process documentation [14] of the licence holder and in the relevant ageing management programs developed for SSCs relevant for the nuclear safety. The licence holder has an ageing management system for SSCs relevant for nuclear safety, aimed at maintaining their design safety functions during intended long-term operation. The ageing management process is implemented on the units of NPP EBO V2, NPP EMO in operation, as well as units of NPP MO3&4 under construction.

For the AM of classified equipment, the licence holder has the required management and implementing documentation. The scope and the system for AM documentation management corresponds to the requirements of ÚJD SR (according to BNS I.7.4/2016 Section 6.4 [66] and BNS I.9.2/2014 par. 5.3.8 [5]). The licence holder has a system of administrative checks in accordance with the requirements of ÚJD SR (according to BNS I.7.4/2016 Section 6.4 and BNS I.9.2/2014 par. 5.3.8). The licence holder reviews operating procedures relating to AM, and has an established validation system in accordance with the requirements of ÚJD SR (according to BNS I.7.4/2016 Section 6.4).

Based on the overall assessment, the strengths of the AM program include:

- In the strategic plans in the field of AM, in the methodology documentation the licence holder addresses not only the issue of obsolescence, but also the development of long-term rehabilitation strategy for SSCs,
- The licence holder maintains a special database for the AM purposes,
- Development of AMP for classified equipment for a long-term operation.

Among identified weaknesses are deficiencies in the SSC drawings in relation to the actual status. Ageing management databases should be updated continuously to reflect the actual status and knowledge.

The overall ageing management program for classified equipment at NPPs in Slovakia can be considered as appropriate. There were no deficiencies identified that would require adopting immediate corrective actions.

3 Electrical cables

3.1 Description of ageing management programs for electrical cables

The ageing management program for cables is implemented by the licence holder and is carried out in accordance with the guide [67]. This guide is valid for all nuclear units in Slovakia, i.e. operating units of NPP EBO V2, NPP EMO and units of NPP MO3&4 under construction. The program is developed in accordance with the national [5] and international guidelines and recommendations and contains the basic attributes of an effective ageing management program [1].

The ageing management program for cables started in the period 2000-2003, when the selection of cables for periodic cable measurements in operation was made for NPP EMO and NPP EBO V2, the methodology was developed for monitoring cable lifetime, and also the surveillance sampling program for NPP EMO [68], [69] and [70].

3.1.1 Scope of ageing management for electrical cables

When developing the program, the procedure for the selection of representative samples from the types of cables used at the power plant was chosen. For selection of cables for the program of ageing management, the following criteria were used primarily:

- Safety aspect – Cables included in the AMP belong to the safety or safety relevant cables, used for electric supply for the consumers (transfer of information for I&C cables) belonging to the Safety Classes 2 and 3 according to [19],
- The environment, in which the cable is located – This criterion is important in relation to the effect of stressors from the surrounding environment, and the resulting degradation mechanism and effect of ageing on the cables. Based on this, the emphasis was put on cables installed in the containment (effect of increased temperature and radiation) and selected areas outside of containment with increased temperature.

Within the AMP, the status of cables and the degree of their degradation are determined by the following sub-programs:

- Functional cable measurements in operation – This program consists in the periodic measurements of selected electric parameters on selected cables in operation to verify the condition of the cables under operating conditions. The measurements are carried out during scheduled outages of NPP units. Under this sub-program, the ageing mechanisms are monitored through changes in the electrical properties of cables. The program was introduced on both NPP EBO V2 and NPP EMO in 2001.
- Surveillance Specimen Program – The program consists of preparation, cable placement and periodic sampling and evaluation of samples of representative types of cables placed in the selected areas of NPP. The samples are placed in the most heat and radiation exposed areas of NPP. Under this sub-program, the ageing mechanisms are monitored through changes in mechanical, thermal and oxidation and electrical properties of cables. The program was introduced on NPP EMO in 1999 and 2001 and at NPP EBO V2 from 2002 till 2004.

The list of cables included in the AMP is given in [15], [16] and [17]. The cables included in the AMP are divided into three main categories in terms of voltage level:

- Cables of voltage level 6 kV – These cables are assessed through the program of function cable measurements in operation. A brief overview of 6 kV cables monitored as part of function cables measurements in operation is given in Table 3-1.

- Cables of voltage level 0.4/1 kV – These cables are assessed through the program of function cable measurements in operation and surveillance specimen program. A brief overview of 0.4/1 kV cables monitored as part of function cables measurements in operation is given in Table 3-2.
- Cables of the instrumentation and control system (I&C) – These cables are assessed through the program of function cable measurements in operation and surveillance specimen program. The primary reason for including I&C cables into AMP is their location in temperature and radiation exposure areas. A brief overview I&C cables monitored as part of function cables measurements in operation is given in Table 3-3.

Table 3-1 Cables of voltage level 6 kV included in the AMP

Powered consumer	Cable routing	Nuclear power plant (number)		
		EBO V2	EMO	MO3&4
Main circulating pump	outside containment/ containment	12	9	6
Feed water pump	outside containment	2	5	6
ESW pump	outside containment/ external ducts	1	10	6
Cooling water pump	outside containment/ external ducts	1	4	4
Diesel generator	outside containment/ external ducts	1	4	4
Emergency system pump	outside containment	1	2	4
Main make-up water pump	outside containment	1	4	2
Emergency make-up water pump	outside containment	-	1	4
Super-emergency feed water pump	outside containment/ external ducts	-	1	2
Transformer	outside containment/ external ducts	-	9	6

Table 3-2 Cables of voltage level 0.4/1 kV included in AMP

Powered consumer	Cable routing	Nuclear power plant (number)		
		EBO V2	EMO	MO3&4
Emergency system pump	outside containment	2	1	2
Safety system valves	outside containment/ containment	5	-	-
Emerg. make-up demi-water pump	outside containment/ external ducts	1		
RSA valve	outside containment/ HELB area	-	6	6
HVAC fan	outside containment/ containment	-	4	4
Electric heater of pressurizer	outside containment/ containment	-	4	6
Pump of spent fuel	outside containment	-	3	2

Powered consumer	Cable routing	Nuclear power plant (number)		
		EBO V2	EMO	MO3&4
pool cooling system				
Intermediate circuit pump	outside containment	-	3	4
Pump for the cool down system	outside containment	-	1	2
Auxiliary circulation pump	outside containment/containment	-	2	2
Boron concentrate pump	outside containment	-	-	2
Organized leakage pump	outside containment	-	-	2

Table 3-3 I&C cables included in AMP

Measurement	Cable routing	Nuclear power plant (number)		
		EBO V2	EMO	MO3&4
Pressure in MSH	outside containment/ HELB area	-	9	12
Coolant pressure in the loops	outside containment/ containment	-	6	6
Pressure in the SG box	outside containment/ containment	-	7	-
Water level in the SG, pressurizer, A201	outside containment/ containment	-	2	8
Temperature of coolant in HCP	outside containment/ containment	-	3	6
Temperature in pressurizer	outside containment/ containment	-		6
Temperature at the outlet from fuel assembly	outside containment/ containment	-	2	2
Other temperature measurements	Outside containment/ containment	-	14	4

Explanation of the item "cable routing":

– Outside containment – LOCA-free areas with a total design dose during its lifetime of less than 100 Gy, and areas outside the external cable ducts,
– Containment – areas with the possibility of creating emergency conditions in the environment, LOCA type,
– External ducts – underground cable ducts between individual buildings,
– HELB – areas outside containment with the possibility of environment emergency conditions of HELB type.

PVC, PE, XPE and EPR are used as cable and jacket insulation materials. Polyvinyl chloride is mainly found on cables in NPP EMO and NPP EBO V2. The remaining materials (PE, XPE and EPR) are used in all power plants.

The list of cable types included in the surveillance specimen program is given in Table 3-4. With the exception of one type of 6 kV cable located in NPP EBO V2, the surveillance specimen program includes signal and control cables of voltage level 0.4/1 kV and I&C cables.

Table 3-4 List of cable types included in the surveillance specimen program

No.	Cable type	EBO unit 3	EBO unit 4	EMO unit 1	EMO unit 2
1	6-CXKFE-R 1x240/LOCA	X	X		
2	AYKY 3Bx4	X			
3	AYKY 4Bx16			X	X
4	CXKE-V/LOCA 12Cx1,5		X		
5	CXKE-V/LOCA 24Cx1,5	X			
6	CXKE-V/LOCA 4Bx1,5		X		
7	CXKE-V/LOCA 7Cx1,5	X			
8	CYKY 3x4	X			
9	CYKY 7Cx1,5			X	X
10	CHKE-V J 4x1,5	X			
11	CHKE-V/LOCA 7Cx1,5	X			
12	CHKH-V180 12Cx2,5		X		
13	JC5XFE-R/LOCA 12x2x0,35		X		
14	JC5XFE-R/LOCA 8x2x0,35		X		
15	JXFE-R 1x2x0,8	X	X		
16	JYTY 7Cx1			X	X
17	JYTY 7Dx1			X	
18	KMPEVEng 37x0,35			X	
19	KMPEVEng 4x0,35			X	X
20	KMTVEV-CHK 8x2,5			X	X
21	KPETI 1x2x0,7			X	
22	KPETI 7x2x0,7	X		X	X
23	KPETI-XA 1x2x0,7	X		X	X
24	KPoBVng 27x1,5			X	
25	KPoBVng 7x2,5			X	X
26	KPoEVng 7x1,5			X	X
27	KPOSG 14x1,5	X			
28	KVVGE 10x1,5	X			

By comparing the range of cables in AMP with the target cable groups defined in TPR Technical specification, it follows that:

- Group 1 in TPR (High voltage cables subject to adverse environment) corresponds to cables of 6 kV group. The 6 kV cable located in the area of containment at NPP EBO V2 and NPP EMO are included in AMP – these are MCP cables. Other 6 kV cables included in the AMP are located outside the containment and in external cable ducts, where the stressors affecting the cables include: ambient temperature, humidity and Ohmic heating. An important factor in selecting these 6 kV cables was the safety aspect.
- Group 2 in TPR (Medium voltage cables concealed or in trenches) corresponds in terms of voltage level, to 0.4/1 kV cables. There are no buried 0.4/1 kV safety cables at the Slovak NPPs. The 0.4/1 kV cables are laid in trays in external cable ducts or in cable trenches and on the trays inside the buildings and are accessible for visual inspections. When selecting 0.4/1 kV cables for AMP the safety aspect was taken into account (power supply for emergency system pumps)

together with the environment where the cables are located (effect of increased temperature and radiation).

- Group 3 – neutron flux measurement cables – is not directly included in the AMP. Group 3 of cables in AMP, in terms of voltage level are I&C cables. The primary reason for including I&C cables into AMP is their location in the most exposed areas of the NPP in terms of temperature and radiation. Cables for measuring neutron flux in the part from the ionization chamber to the transfer block, at all NPPs are made with mineral insulation, i.e. they do not contain polymeric materials and monitoring of ageing effects is not justified.

Individual sub-programs implemented in AMP, the whole cable or its sub-components, are monitored in a comprehensive manner. Assignment of cable elements by sub-programs within AMP, is shown in Table 3-5. A detailed description of activities and tests carried out on cable elements within the sub-programs is given in par. 3.1.3.

Table 3-5 Coverage of cable elements by AMP activities

Cable element	Function cables measuring in operation	Surveillance specimen program
Conductor	YES	NO
Insulation	YES	YES
Armour/shielding	NO	NO
Jacket	YES	YES
Route termination	YES	NO
The whole cable	YES	YES

Within the cable AMP, also monitoring of parameters of the environment is carried out, to which the cables are exposed in operation. Monitoring includes measurement of temperature, radiation dose and relative humidity in selected areas of containment and also outside containment (piping space in the electrical building +14.7 m) at both power plants in operation. The basic objectives of monitoring include verification of design parameter values of the environment, and identifying of hot spots. Information obtained from monitoring is important for the feedback; i.e. qualified life of the cable can be corrected according to the current conditions of the environment and by identifying hot spots it is possible to take preventive measures to mitigate the effects of ageing or the relevant corrective actions.

3.1.2 Assessment of ageing of electric cables

The cable life is generally limited by the properties of insulating materials of conductors and cable jackets.

Cables operated under power plant conditions are exposed to the effect of stressors, resulting in a change of material properties of individual cable elements. The most important stressor is the effect of parameters of the environment on the insulating parts of the cable, which are the limiting component of the cable functionality, not only under normal operating conditions, but especially during emergency conditions, when there is a functionality requirement for safety relevant cables.

The temperature and radiation are most affecting the cables from among ambient parameters. These parameters are quantified in the context of monitoring the environment (see par. 3.1.1). For the 6 kV power cables, another contributing factor is the Ohmic heating due to current load. By temperature and radiation, the physical and chemical processes occur in the polymeric materials of the cables, which at the macroscopic level are manifested mainly by the change of elastic properties of the insulating materials, i.e. loss of elasticity – embrittlement. These changes in elastic properties are assessed by measuring the strength and tensile strength of the polymeric insulation materials of the cable (conductor insulation and jacket). Measurement of elongation at break is performed within the cable surveillance specimen program. The test is standardized, verified and world-wide regarded as the main method for assessing degradation of polymeric materials. The acceptance criterion for measuring the elongation of cables in operation is set to ≥50% of the absolute value. In the cable AMP, this value represents the main criterion for the assessment of cable ageing and also represents the limit value, below which the cable is considered unsuitable. Selection of the acceptance criterion was based on good practice and the IAEA recommendations [71], [72]. Newly installed cables must meet the values of elongation at break as prescribed by the manufacturer.

Another stressor affecting the cable is moisture. Increased humidity is the factor, to which the cables placed mainly in the outer cable ducts are exposed to. The effect of moisture is manifested primarily by reduction of the dielectric strength and insulation condition of the cable. These changes are evaluated within the AMP by measuring electrical parameters – insulation resistance, polarization index, loss factor, and reflectometric TDR measurements. The measurement of electrical parameters is mainly applied to cables installed in operation (ECAD measuring system), as well as to the surveillance specimen program. Internal regulations, as well as international documents [73], [74] were taken into account when selecting acceptance criteria for electrical measurements within the PSR.

A summary of the main identified ageing mechanisms within the AMP is given in Table 3-6.

Table 3-6 Summaries of the main ageing mechanisms within AMP

Ageing mechanism	Stressor	Effect of ageing	Part of cable	Sub-program within AMP	Applied monitoring method
Temperature and Radiation embrittlement	temperature (impact of environment + ohmic heat.) dose input	cracking	insulation jacket	surveillance specimen program	measuring elongation
		decrease in dielectric strength increase in leakage currents		measuring function cables in operation surveillance specimen program	measuring electrical properties
Moisture penetration	moisture	decrease in dielectric strength increase in leakage currents	insulation jacket whole cable	measuring function cables in operation	measuring electrical properties

The licence holder participates in the following international projects in the field of cable ageing management:

- CADAK (Cable Ageing Data and Knowledge) – an international project managed by OECD/NEA, which is aimed at sharing and exchange of information and results of research in the field of assessment of cable ageing and qualification,
- IGALL (International Generic Ageing Lessons Learned),
- International Equipment Reliability Working Group.

A supplier participating in ageing management and cable qualification, was one of the parties of the IAEA project “Coordinated Research Program on Qualification, Condition Monitoring and

Management of Ageing of Low Voltage Cables in Nuclear Power Plant Life Management“, 2012-2015. The main objective of the projects was to identify monitoring methods with the potential for monitoring degradation and assessment of ageing of different types of insulation materials.

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

In line with the recommended format of the report, the description of activities performed on the cables within AMP is described by cable elements and cable groups (NAR example) as shown in par. 3.1.1.

3.1.3.1 Element – Conductor

Table 3-7 Activities performed for the element – conductor

Cable group	Measurement of functional cables in operation	Surveillance specimen program	Other activities
6 kV cables	measuring DC resistance, TDR	-	-
0.4/1 kV cables	measuring DC resistance, TDR	-	-
I&C cables	measuring DC resistance, TDR	-	-

As discussed in par. 3.1.2, the ageing mechanisms having effect on the service life and functionality of cables, are not foreseen for the conductors. In the course of these activities, the conductor is monitored primarily for its continuity (integrity), DC resistance and for open ends also their appearance. Within measurements of functional cables in operation, measuring DC resistance and conductor continuity is evaluated by TDR measurements.

The test frequency is given by the schedule of individual activities – see par. 3.1.3.6.

Acceptance criteria for the tests specifically focusing on conductors are not set.

3.1.3.2 Element – Insulation

In terms of potential degradation and effects of ageing, the conductor insulation represents the most important element that affects cable life and functionality. Within measurements of functional cables in operation, measuring of electrical parameters is performed, i.e. insulation resistance, polarization index, loss factor and TDR. These measurements are not specifically performed on the insulation only, but on the whole cable in the configurations “conductor/conductor“, and “conductor/earth“. It means that they include also conductor insulation and therefore are listed here. The surveillance specimen program, besides the measurement of electrical parameters (insulation resistance, polarization index, loss factor) includes also measurements of mechanical properties (elongation, tensile strength) and thermic-oxidation characteristics (time/temperature of oxidation induction). From this set of tests, the measuring of insulation elongation at break is the most representative parameter to monitor insulation degradation due to external stressors.

In addition to the AMP, also measuring of insulation condition of cables is performed as part of technical inspections, testing and certification, and their extent and frequency is given by the requirements of national legislation [75].

The test frequency is given by the schedule timetable of individual activities. Measurements of functional cables in operation are performed in NPP EBO V2 at 1 year interval (once in 3 years for MCP) and in NPP EMO at interval once in 3 years. The sampling frequency for surveillance specimens for the units of NPP EBO V2 once every 4 years and for units of NPP EMO once every 2 years, while the relevant type of cable (not the design) shall be assessed at least once every 5 years. Higher sampling frequency for NPP EMO is given by a large number of samples of material compared to NPP EBO V2.

Table 3-8 Activities performed for the element – insulation

Cable group	Measurement of functional cables in operation	Surveillance specimen program	Other activities
6 kV cables	Riz, polarization index, loss factor, TDR	elongation, tensile strength, Riz, loss factor, OIT(p)	technical inspections, testing and certification
0.4/1 kV cables	Riz, polarization index, loss factor, TDR	elongation, tensile strength, Riz, loss factor, OIT(p)	technical inspections, testing and certification
I&C cables	Riz, polarization index, loss factor, TDR	elongation, tensile strength, Riz, loss factor, OIT(p)	-

Acceptance criteria for the tests specifically focusing on conductor insulation are as follows:

- Measurements of functional cables in operation – see par. 3.1.3.6,
- Surveillance specimen program,
  - Elongation at break test - min. 50% absolute value
  - For electric parameters, see par. 3.1.3.6.

Other tests (tensile strength and OIT(p)) are performed as complementary tests and their acceptance criteria are not defined.

3.1.3.3 Element – Armouring/ shielding

Armouring and shielding of the cable are not specifically monitored within the AMP, as they are made of metallic materials and monitoring of ageing effects has no justification. For proper shielding functionality, as part of maintenance activities its proper grounding is monitored.

3.1.3.4 Element – jacket

Table 3-9 Coverage of cable elements by AMP – jacket

Cable group	Measuring functional cables in operation	Surveillance specimen program	Other activities
6 kV cables	Riz, polarization index, loss factor, TDR	elongation, tensile strength, Riz, loss factor, OIT(p)	technical inspections, testing and certification
0.4/1 kV cables	Riz, polarization index, loss factor, TDR	elongation, tensile strength, Riz, loss factor, OIT(p)	technical inspections, testing and certification
I&C cables	Riz, polarization index, loss factor, TDR	elongation, tensile strength, Riz, loss factor, OIT(p)	-

The condition of cable jacket is primarily assessed in the surveillance specimen program by measuring of mechanical properties (elongation at break, tensile strength). In addition, on surveillance specimens the thermal-oxidation characteristics are also monitored (time/temperature to oxidation induction) and electrical properties. Measurement of the functional cables in operation is carried out by measuring the electrical parameters, i.e. insulation resistance, polarization index, loss factor and TDR. These measurements are carried out on the entire cable, in "conductor/ earth" configurations. It follows that they also include the jacket of the cable and are therefore listed here. Outside of the AMP, measuring of the cable insulation condition is performed within technical inspections, testing and certification, where the range and the frequency are determined by the requirements of the national legislation [75].

The test frequency as well as the acceptance criteria is the same as in par. 3.1.3.2.

3.1.3.5 Element – Route termination

Table 3-10 Coverage of cable elements by AMP – route termination

Cable group	Measuring functional cables in operation	Surveillance specimen Program	Other activities
6 kV cables	measuring DC resistance	-	maintenance
0.4/1 kV cables	measuring DC resistance	-	maintenance
I&C cables	measuring DC resistance	-	-

Cable route termination is monitored indirectly within AMP by means of measuring DC resistance of the entire cable route (including transient cabinets) and the connected consumers.

Outside the AMP, during maintenance on equipment within the EQR process, a visual inspection of cable connections in the switchboard and transient cabinets is performed according to the relevant maintenance templates, and on selected terminals also thermo-vision measurements.

The test frequency is given by the schedule for individual activities – see par. 3.1.3.6. Activities under EQR are realized according to SMS.

3.1.3.6 Element – The entire cable

Table 3-11 Coverage of cable elements by AMP – the entire cable

Cable group	Measuring functional cables in operation	Surveillance specimen Program	Other activities
6 kV cables	Riz, polarization index, loss factor, TDR	Riz, polarization index, loss factor	technical inspections, testing and certification
0.4/1 kV cables	Riz, polarization index, loss factor, TDR	Riz, polarization index, loss factor	technical inspections, testing and certification
I&C cables	Riz, polarization index, loss factor, TDR	Riz, polarization index, loss factor	calibration of resistance measuring

The condition of the entire cable is monitored by measuring its electrical properties. Measuring of functional cables in operation includes also measuring of insulation resistance, polarization index, loss factor and TDR measurement. These measurements are carried out in configurations "cable to cable“, and “cable to earth“. Electrical values (insulation resistance, polarization index, loss factor) are also monitored in the surveillance specimen program, where the whole length of the sample is measured, i.e. min. 6 m.

Outside of the AMP, measuring of the insulation condition is performed on 6 kV and 0.4/1 kV cables within technical inspections, testing and certification, where the range and the frequency are determined by the requirements of the national legislation [75]. The condition of selected I&C cables that are part of the measuring circuit, is monitored as part of regular calibrations and verifications of measuring circuit.

The test frequency is given by the schedule for individual activities. Measurements of functional cables in operation are conducted at NPP EBO V2 at an interval of once a year (once every 3 years for MCP) and at NPP EMO at an interval once every 3 years. The sampling frequency for the units of NPP EBO V2 is once every 4 years and for the units of NPP EMO once every 2 years, while the appropriate type of cable (not the design) is assessed at least every 5 years. Higher sampling rate at the NPP EMO is given by a larger number of samples of material when compared to NPP EBO V2.

Acceptance criteria for the tests specifically focusing on conductor insulation are:

- Measurements of functional cables in operation,
    - Insulation resistance
        - For 0.4/1 kV cables – according to EPRI document [74]
        - For 6 kV cables – according to the internal document of SE [73]
    - Polarization ratio – more than 1, calculated from Riz values at 60 sec and 15 sec,
    - Loss factor ≤ 0.3 (valid for PVC cables).

  For the TDR measurement, the acceptance criterion is not stated, the comparison is made with the previous measurement.
- Surveillance specimen program,
    - Insulation resistance
        - For the new cables – according to the technical conditions of the manufacturer
        - For aged cables 1 MOhm.km
    - Polarization ratio – more than **1**, calculated from Riz values at 60 sec and 15 sec.

3.1.4 Preventive and corrective actions for electrical cables

In accordance with the guide [67] the following corrective actions are applied for individual sub-activities of the ageing management program:

- Measurements of functional cables in operation:
    - Shortening the interval between measurements – if the value of status indicator (insulation resistance, polarization ratio) shows a large parameter change

      This mechanism was applied in the AMP to the cooling water pump cable at NPP EMO, where after a significant drop in the insulation resistance, the original interval was shortened from three years to one year.
    - Cable repair/replacement if the value of status indicator does not meet the acceptability criterion

      This mechanism has not yet been applied in AMP.
    - Visual inspection of the route in case of detecting abnormalities during TDR

      This mechanism has not yet been applied in AMP.
    - Expanding the measured quantities by another that better detects the ageing mechanism,

This mechanism has not yet been applied in AMP. In cooperation with the organization implementing these activities, parallel measurements were made on the selected cables by existing system and system utilizing the principle of linear resonance analysis.

- Extending (decreasing) the range of cables if there is a location in the given space or increased degradation was observed on the given cable type,

  This mechanism was applied at NPP EMO, where the range of cables was extended by cables for RSA. This valve, as well as a large part of the route is located in an area with high temperature load, where increased degradation was observed.
- Surveillance Specimen Program,
  - Shortening the interval between sampling and evaluation of samples in case:
    - If the values of status indicator (insulation resistance, elongation) measured on the sample are worse than the values measured during the qualification process
    - If the values of status indicator (insulation resistance, elongation) measured on the sample show a rapid negative change

    This mechanism has not yet been applied in the AMP.
  - Extending the interval between sampling and evaluation of surveillance specimens in case the values of the status indicator (insulation resistance, elongation) measured on the sample do not show any change or show only negligible change

    This mechanism was applied on surveillance specimen at NPP EBO V2, after the test results did not show significant change in parameters and due to the adequacy of cable material for the needs of long-term operation. Interval of sampling was extended from two years to four years
  - Expanding the measured values by another one that better detects the ageing mechanism.

    This mechanism was applied in such a way that during the program, the existing electrical and mechanical tests were extended by thermic-oxidation tests – measuring time/temperature to oxidation induction
  - Expanding the scope of surveillance specimens

    This mechanism is applied continuously if the cables are replaced mainly in the containment. As an example, the surveillance specimen program at NPP EBO V2 was extended by a cable at MCP (primary side from the containment penetration to the engine), which was installed in 2015 and 2016 as a replacement for the original cable.
  - If the sample does not meet the required acceptability criterion, a checking measurement on another sample should be performed; if this checking measurement confirms the original results, then a cable replacement recommendation should be issued.

    This mechanism was applied at NPP EBO V2 in 2005, when an inadequate value of elongation at break of cable insulation was found in the initial measurement of surveillance specimen of the newly installed cable. The cable was then replaced with a new one with compliant parameters.

If the value of status indicator does not meet the acceptability criterion, then the established procedure for the Correction and Prevention System is applied, i.e. an “NG notification” is filled in.

3.2 Experience of the licence holder with the application of AMP for the electrical cables

The licence holder started to use the cable ageing management program at NPP EBO V2 and NPP EMO in the period 1999-2001. During the implementation of the cable ageing management program, there were a number of changes resulting from the internal experience and needs of the licence holder or from the latest trends in cable condition monitoring. Examples of such changes applied in the AMP are given in par. 3.1.4.

By participating in CADAK and IGALL international projects, the licence holder maintains access to the latest knowledge and trends in the field of ageing management and cable lifetime assessment.

Through individual partial activities within the cable AMP (surveillance specimen program, measurement of functional cables in operation) the licence holder covers the main degradation mechanisms identified based on operational experience and international recommendations.

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

The licence holder implements the cable ageing management program in accordance with the guide [67]. This guide is valid for all nuclear units in Slovakia, i.e. NPP EBO V2, NPP EMO in operation, and units of NPP MO3&4 under construction. Through individual sub-programs of the cable AMP (surveillance specimen program, measurement of functional cables in operation) the licence holder covers the main degradation mechanisms identified based on operational experience and international standards. The licence holder monitors also parameters of the environment (temperature, radiation dose, relative humidity), to which cables in operation are exposed. Monitoring includes the containment area and also the area outside the containment at both power plants in operation.

After refuelling of each operating unit and before its start-up, the results of in-service inspections of classified equipment are assessed with the participation of ÚJD SR. Thereafter the licence holder sends reports to the ÚJD SR from the preliminary assessment of results, and from assessment of results of in-service inspections of classified equipment in a form of regular reports in accordance with the ÚJD SR´s Decision [3] and the ÚJD SR issues its opinion on these. In-service inspections of classified equipment are performed by the licence holder according to the programs of in-service inspections approved by the ÚJD SR. The licence holder is obliged, in accordance with the Decree of the Ministry of Labour, Social Affairs and Family No. 508/2009 Coll., to perform inspections, testing and certification of electrical technical equipment. In the course of planned ÚJD SR inspections aimed at checking the implementation of in-service inspections, inspections, testing and certification of electrical equipment and inspections after refuelling, the ÚJD SR inspectors by random selection check also the documentation prescribed by ageing management programs for power cables and the actual condition of power cables. In addition to planned inspections as mentioned above, the ÚJD SR conducts also unplanned inspections in response to various events in the world. Ageing management is subject of inspections on periodic safety review.

In most cases, the inspections were without any findings. In case of a finding, the ÚJD SR issues a Protocol to the licence holder, containing corrective actions and deadlines for their fulfilment, and then checks fulfilment of these corrective actions within the set deadlines. No major deficiencies were identified during these checks that would require immediate corrective action. The capability of systems, structures and components relevant for nuclear safety to fulfil their safety functions, is secured.

4 Concealed pipework

4.1 Description of ageing management programs for concealed pipework

4.1.1 Scope of ageing management for concealed pipework

Ageing management of concealed pipes of the licence holder covers underground ESW pipes, for which an ageing management program is implemented [76]. The program is developed according to national [5] and international guides and recommendations, and contains the basic attributes of an effective ageing management program [1]. This guide is valid for NPP EBO V2 and NPP EMO in operation.

The activities to manage ageing of concealed ESW pipes started in NPP EMO from the start of operation within the program of in-service inspections, and in NPP EBO V2 in 2008.

4.1.1.1 Identification of concealed pipes

Identification of concealed pipes was considered for underground pipes. In NPP EBO V2, NPP EMO and NPP MO3&4 there are different concealed technological and non-technological piping systems that are different in terms of their safety relevance, method of their laying in the ground and the availability for monitoring. Not included are: wastewater piping from the operation NPP EBO V2, NPP EMO and NPP MO3&4, with checked content of radioactive substances, as well as concealed pipes that are not underground, but located in the controlled area in the main generation block.
None of the piping routes in SE located in the ground contains radioactive substances.

Concealed/ Buried pipes in the ground are part of the systems stated below.

Essential Service Water
Essential service water pipes are safety relevant steel pipes that at NPP EBO V2 are largely placed in a concrete block. In the sections between the fan cooling towers (FCT) and the central pumping station (CPS), the pipes are buried in the ground and are protected from the outside by protective insulation and active cathode protection, and in sections near CPS and FCT the segments are above ground and are protected from the outside by protective coating and thermal insulation, under which resistance heating is installed. The ESW pipes to DGS and HPCS are guided through passage pipe ducts and from outside they are protected by protective coating.
In NPP EMO and NPP MO3&4 the ESW piping is placed in passage pipe ducts and from outside they are protected by protective coating.

Super-emergency feeding of steam generators
Pump discharge pipes of the super-emergency feeding are safety relevant steel pipes, protected from the outside by protective coating, and partly are guided through passage and partly through semi-passage, seismically resistant ducts. In NPP EBO V2 the pipes are additionally protected by thermal insulation.

Fire water
At NPP EBO V2 the fire water pipes are not part of the seismic concept. In the period 1993-2000 NPP EBO V2 implemented in four phases replacement of the original metal pipeline for PVC pipe. The pipes are partly buried and partly placed in passage pipe duct.
In NPP EMO and NPP MO3&4 part of the piping is seismically resistant. These pipes are made of steel and are guided in passage, seismically resistant pipe ducts. From the outside they are protected

by a protective coating. Pipes that are not part of the seismic concept are plastic and are placed in the ground.

Circulating cooling water
Circulating cooling water pipes are steel pipes that in NPP EBO V2 are largely placed in a concrete block, except for the sections in front of the cooling towers and at CPS, where they are buried in the ground and are protected from outside by a protective coating. In NPP EMO and NPP MO3&4, the CCW pipes are placed similarly as in NPP EBO V2. They are also placed in a concrete block, except for the short sections of the connection to the pump station building, where they are buried in the ground and from the outside they are protected by a protective coating.

Decarbonized water to make up the losses in CCW and the CCW blowdown
Piping of decarbonized water for making up the losses in CCW and the CCW blowdown piping are steel pipes that in NPP EBO V2 are buried in the ground and from outside protected by protective coating. In NPP EMO and NPP MO3&4 there is steel piping placed in a concrete block parallel to the CCW piping.

Makeup water for the refill of the ESW system
The makeup water piping for the refill of the ESW system is a steel pipe that in the NPP EBO V2 is buried in the ground and from outside protected by protective coating. At NPP EMO and NPP MO3&4 the steel pipe is guided through pipeline bridges and through passage ducts and has an external protective coating. The ESW blowdown in the NPP EMO is in the ground and is fed into the inlet object of CCW pumping station (before the suction of cooling water pumps).

Service water (non-essential)
The piping of the service water (NESW) in NPP EBO V2 is largely placed in a concrete block, except of the above-ground section near CPS, which is protected from the weather conditions by coating, heating and thermal insulation. The NESW piping in NPP EMO and NPP MO3&4 are similarly placed in a concrete block, except the routes for refilling to the CCW building, which is placed in the ground and has an external protective coating.

Diesel oil
Piping for transporting fuel from the fuel management to DGS is made of steel, which in NPP EBO V2 placed in passage and non-passage pipe ducts. In NPP EMO and NPP MO3&4 the steel piping for diesel oil is placed in non-passage ducts and has protective coating.

Hydrogen and nitrogen
In 2016, in NPP EBO V2 the hydrogen management distribution piping was replaced, the piping routes in the ground and piping distributions to the turbine hall to individual generators. The original two pipeline routes between the hydrogen reduction station and the turbine hall were replaced with new plastic pipelines. Pipes were buried in a sand bed, covered with insulation and covered by soil. Crossings of the plastic pipes under the road are through concrete penetrations. The hydrogen distribution in the turbine hall and in the reduction station itself is made of stainless steel. The nitrogen distribution piping in NPP EBO V2 is placed in pipe ducts.
Piping for distribution of hydrogen and nitrogen in NPP EMO is placed in non-passage piping ducts, equipped with insulation. Hydrogen piping is buried in sand for increased safety in the entire cross-section of the duct.

4.1.1.2 Concealed pipes covered by ageing management program

The ageing management program for the concealed pipes in SE, a. s., focuses on concealed ESW pipes. In order to ensure operability of the ESW pipes during the design lifetime, as well as during the planned long-term operation of nuclear units of SE, a. s., the ageing management program for the essential service water pipes [76] was developed and is common for NPP EBO V2 and NPP EMO and NPP MO3&4.

Program addresses ageing management of external and internal connecting ESW piping.

According to the ÚJD Decree [19] the ESW system belongs to classified equipment and is categorized as safety class BT III: "heat removal from the safety systems up to the first accumulation volume sufficient to perform safety functions in addition to the basic heat removal systems included in the safety class II, par. d) and e)".

The ESW system is designed as redundant system, 3 times 100%, while in NPP EBO V2 each system has 2 pumps per unit (2 times 100%) and 2 sections FCT (2 times 100%), and in NPP EMO and NPP MO3&4 there are 3 pumps per unit (3 times 100%) for each system, and 2 sections FCT (2 times 100%). The ESW system is resistant against single failure (one single failure of active equipment per unit or one single failure of pressurized passive equipment on both units) and common cause failure (fire, flooding, seismic effects, interaction from high-energy pipes, airborne debris, load crash, ambient conditions and extreme climatic conditions). Parts of the ESW circuits are physically separated from each other. Each ESW system is powered from another section of secured power supply of category II in accordance with the power supply of independent circuits of the emergency core cooling.

The ESW system is common for both units of individual NPPs in the part pools, ESW pumps and the main piping distribution.

In the turbine hall the piping distributions for each system are branched into individual units, separated by unit closing valves. Piping routes for cooling the diesel generators are in NPP EBO V2 connected to the main ESW route in the turbine hall and in NPP EMO and NPP MO3&4 in the piping ducts.

The ESW outer connecting pipe provides for transport of ESW:

- From the fan cooling tower pools to suction pits of ESW pumps in CPS,
- From the ESW pumping station to the consumer devices in the turbine hall and reactor hall – supply pipe,
- From consumer devices to the fan cooling towers – return pipe.

External ESW connection pipe consist of 3 independent ESW systems, structured in the same was in NPP EBO V2, NPP EMO as well as NPP MO3&4 as separate routes according to the operating temperature and pressure:

- Supply lines of the discharge pipes from the ESW pumps in CPS to the turbine hall,
- Supply and return pipes to the diesel generator station (DGS) and high-pressure compressor station (HPCS),
- Supply and return pipes for the nitrogen management building,
- Return pipes to FCT,
- Return gravity piping from FCT to CPS.

External connecting pipe at NPP EBO V2

The supply and return ESW piping routes between CPS and the turbine hall, together with NESW piping and circulating water piping are mostly embedded in a concrete block buried in the ground. The section of the supply piping between CPS and the concrete block and the section of the return pipe between the unit and FCT, were originally buried in the ground. After the modification of these sections the supply pipe exits from CPS above ground and enters the inspection shaft, where the pipeline is connected to the pipeline in the concrete block. The ESW return pipe exits at FCT No. 582c from the monolith to the inspection shaft, from which the ESW system 1 continues freely laid in the ground to FCT No. 582a, and ESW systems 2 and 3 exit from the shaft to the surface and continue above ground to FCT No. 582b and FCT No. 582c. The pipeline in the shaft and the above ground piping is protected by protective coating and thermal insulation, under which resistance heating is installed. Piping of system 1, freely laying in the ground, is protected by outer PE insulation. Return (gravity) ESW piping in the section from FCT to CPS is freely covered in the ground at a depth of 3 to 7 m, and is protected from the outside by viscous – elastic insulation STOPAQ and cathode protection.

External connecting piping in NPP EMO and NPP MO3&4

Piping system in the NPP EMO and NPP MO3&4, the same way as in NPP EBO V2, divided into three separate systems and to separate sections according to their operating temperature and pressure. With regard to the category of seismic resistance 1b they are located in seismically resistant passage piping ducts.

4.1.2 Assessment of concealed pipework ageing

The assessment of concealed pipes in SE is systematic for the ESW pipes, for which SE has an AMP [76]. The ageing management program contains parameters of piping routes of ESW systems, as well as the characteristics of components of the piping routes. The results of component wall thickness measurements and results of measuring mass loss of coupon weight in corrosion loops are an output from AMP.

The assessment of buried ESW pipes follows the methodology implemented in the EPRI software BPWORKS™, which was introduced as part of AMP in NPP EBO V2 and NPP EMO in 2016. The database section of the program included data on the ESW pipes, piping insulation, soil, design parameters of the routes and individual components. In the “Risk Assessment” program module, the risk of degradation of individual sections of piping was subsequently determined. The assessment was realized for ESW piping in NPP EBO V2 and also NPP EMO.

Understanding ageing is based on knowledge of the following data:

- Materials and material properties,
- Operating load,
- Ageing mechanisms,
- Degradation locations,
- Indicator of degradation degree,
- Consequences of degradation by ageing and subsequent failures during normal and abnormal operation.

4.1.2.1 Identification of degradation mechanisms

The dominant degradation process that limits the life of concealed pipes is corrosion. Its adverse effects increase with the aggressiveness of the external environment and the impact of chemical effects of the service medium. An important role, apart from the type of material, is the voltage level and production technology, mainly the quality of surface layers of the pipe components. Great impact on the life of buried pipes has inconsistent compliance with the quality of performed installation work – damage to the material, not using sub-base material, tolerance of leakages of surrounding pipes, improper workmanship of insulation works.

The inner surface of concealed pipes is affected by:
- Uniform/general/corrosion,
- Pitting corrosion.

The outer surface of concealed pipes is affected by:
- Atmospheric corrosion mainly of general character,
- Microbiological corrosion – for freely laid pipes,
- Eddy currents - for freely laid pipes.

4.1.2.2 Assessment of ESW pipe ageing

The range of monitored systems, structures and components (SSC) within the ESW AMP [76] in NPP EBO V2 is stated in [77], [78] and for NPP EMO [79], [80].

The status indicators are determined to assess the condition of individual SSCs. An equipment status indicator is a defined feature that can be monitored or measured and serves for the estimate or direct indication of the current or future capability of the equipment or parts thereof, to fulfil its function within acceptable criteria.

The status of the ESW piping is assessed on the basis of:
- Selected chemical parameters (Fe, Zn, pH),
- Visual inspections of the ESW piping routes,
- Measurement of mass coupon weight loss in corrosion loops,
- Measurement of ESW piping wall thickness,
- The condition of the concrete protection.

The results of measurements are information about:
- Corrosion rate in mm/year as a difference between the nominal (or the first measured) and last measured thickness related to the given period (from the start of operation or the date of the last measurement),
- Residual life of assessed component.

Based on these data, the components are proposed for replacement (if the lifetime is less than 2 years) or for inspection (if the lifetime is less than 5 years).

The output from AMP for ESW is information about the residual life of the assessed component, corrosion loss and suggestions for measuring and replacing components. This information is part of the Annual Assessment Report on the status of SSCs and the Health Reports.

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

4.1.3.1 Monitoring ESW in NPP EBO V2

In the past, the ESW pipes were checked at NPP EBO V2 in the inspection trenches and at the points of leakages. Following the modification of ESW piping in 2015 and 2016, the monitoring is differentiated depending on the way the pipe is laid.

Rehabilitated ESW piping laid in the ground are actively protected by cathode protection, so monitoring is based on data monitoring from the system of cathode protection. In addition, the pipes are protected by a new waterproofing.

Replaced ESW pipes are available for monitoring in two shafts – where the pipe enters the concrete monolith at the CPS and where the pipe exits the monolith at FCT. Parts of replaced pipes that are led above the ground are inspected visually. After removal of the protective sheet insulation and protective PE insulation, the pipeline is accessible also for manual ultrasonic wall thickness measurements. The measuring points and the inspection interval are shown in IQAP. Detailed mapping of the wall thickness on a small surface area of the component to monitor pitting corrosion can be accomplished using automated ultrasound test system. Implementation of measurement using automated test system is demanding in terms of preparation of the surface area for measurement. It consists of cleaning the entire surface area to pure metal without colour, surface corrosion, sediments or other impurities.

Pipes embedded in a concrete monolith are indirectly monitored by geophysical monitoring of the concrete monolith, which is performed in regular intervals. Within this monitoring, there are geo-radar measurements to map the condition of the concrete monolith and measuring eddy currents.

4.1.3.2 Monitoring of ESW in NPP EMO and in NPP MO3&4

In NPP EMO and NPP MO3&4, inspections of the outer surface of ESW pipes laid in the ducts are performed. The monitoring is carried out by manual measurement of wall thickness by the ultrasonic method at the points of the relevant measuring grid.

The initial selection of points to make wall thickness measurements was done on the basis of IQAP. Based on the experience from the first years of measurements, this selection of components for measurements has been updated on a regular basis. Wall thickness measurement is performed on:

- Elbow segment, where the voltage value exceed 75% of the permissible voltage,
- Other elbow segments and straight sections selected on the basis of engineering experience.

4.1.3.3 Visual inspections

During visual inspections, the status of individual structures, parts of components, weld joints with adjacent portion of the base material and their defects are briefly checked, if apparent on the inspected surface.

Visual inspection is used to monitor the following parameters:

- Damage to outer protective coatings caused by corrosion (bulging), or mechanically,
- Presence of traces of leaking liquids,
- Mechanical damage to piping and fittings,

- Local corrosion damage.

The NDT technician shall issue a “Protocol on visual inspection” about the visual inspection. The Protocol on visual inspection with a result “Non-compliant” is forwarded by the NDT unit to the group for life-cycle management. The specialist for life-cycle management records the indications into DRS.

4.1.3.4 Non-destructive tests – pipe wall thickness measurement

Non-Destructive Tests – ultrasonic pipe wall thickness measurement – are applied at designated locations and in the number as proposed by the group for life-cycle management and approved by the component engineer based on the results of previous measurements and recommendations of IQAP (individual quality assurance program).

The thickness of the pipe wall is measured, while the measured wall thickness of the pipe must not be less than the minimum allowable wall thickness of the pipe.

From the difference between the actual wall thickness and the nominal thickness according to the design by means of rate of corrosion loss (or from coupon corrosion rate) the residual service life of the ESW piping or its individual components, is determined.

On the basis of calculated value of residual life, further procedure for the ESW ageing management is determined by choosing the appropriate chemical protection or by planning the exchange of critical components.

The criteria for determining the minimum permissible wall thickness of the pipe are based on the requirement to ensure reliability of system operation in all conditions considered in the design. They include loss of material due to corrosion and maximum manufacturing tolerance.

Components with a diameter Dn <50 mm are usually not inspected by using ultrasonic method, due to the small contact surface of the ultrasonic probe with the surface of the component, which does not allow stable positioning of the probe on the surface to be checked. The NDT technician issues “Protocol on ultrasonic test“. The Protocol on ultrasonic test shall be forwarded by the NDT unit to the group for life-cycle management. The specialist for life-cycle management records the indications into DRS.

4.1.3.5 Analysis of the influence of chemical composition on the ageing mechanism of ESW pipes

In the process of monitoring influence of the medium on the corrosion of the internal walls of the pipes, an important indicator is the Fe content released by the corrosion process into the medium being transported. The analysis and evaluation of the Fe, Cl content provides a current picture about the state of corrosive environment and its effects on the material to be monitored. Detailed description of all analysed chemical parameters is given in the procedure [81] for NPP EMO or [82] for NPP EBO V2.

Sampling, analysis and evaluation are performed by the department of chemical checking periodically at intervals stipulated by the procedure [81] in NPP EMO and [82] in NPP EBO V2. These are recorded to the CHEMIS system.

For monitoring the corrosion process, in the ESW system there are corrosion loops, in which samples (corrosion coupons) are embedded made from carbon steel. The corrosion rate (even corrosion) is determined from the loss of mass of the corrosion coupons over a given period of time when they are placed in a loop. The loss of mass is converted by calculation to wall thickness decrease. The rate of corrosion loss is given in mm per year. In NPP EMO the system is located on the side filtration of ESW pumping station. In NPP EBO V2 the system is placed in the turbine hall on the ESW feeding piping from the CPS.

4.1.3.6 Monitoring of microbial stimulated corrosion

It is monitored by analysing the deposits and monitoring of coupons (test coupons), evaluation is visual, by means of mass loss, metallo-graphically on cross sections.

4.1.4 Preventive and corrective action for concealed pipework

Corrective actions are based on findings from visual inspections, wall thickness measurements and chemical composition of the flowing medium.

Systemic measures to improve the condition are:
- Component replacement,
- Modification of component design or of the route geometry,
- Change in the material used to manufacture the component,
- Changing operating parameters,
- Application of protective coating system,
- Application of cathode protection system for underground pipes.

Corrective measures to maintain the design status are:
- Repair of damaged insulation of piping laid in the ground,
- Repair of damaged coating on the outer surface of the pipe,
- Removing rust from the outer surface of the pipe,
- Repair of component by applying external build-on welding at the point of reduced wall thickness,
- Repair of component by applying cold and hot sleeves to damaged areas or areas with reduced wall thickness,
- Replacement of component with reduced wall thickness, if one of the repair methods cannot be applied or cannot be effectively applied,
- Adjusting the range of measurements based on actual corrosion losses,
- Repair of fittings – hinges and supports,
- Adjusting the chemical regime, consisting of protection of the piping by checking dosing of conditioners to reduce corrosion rate.

Selection of chemicals, dosing and check of water chemistry is the responsibility of the chemistry department.

4.2 Experience of the licence holder with the application of AMP for concealed pipework

The ageing management program for the ESW was introduced in the period 2008-2011.

At NPP EBO V2, a mapping of the condition of external pipelines was carried out in 2010 and 2011, which revealed inappropriate method of laying the piping in the ground, poor condition of the waterproofing and development of pitting corrosion at places with impaired waterproofing. In 2013, in the framework of construction and technical survey of the concrete monolith and pipes laid in the monolith, a geo-radar survey was carried out in a place with concrete monolith. This inspection showed that there are no significant defects, faults or anomalies on the concrete structure, and the uncovered surface of the ESW piping (after cutting the concrete casing) was free from defect and faults. In 2015, another geo-radar survey was conducted.

Based on the monitoring of the condition, reconstruction of the ESW piping was implemented at NPP EBO V2 in the following range:

- Project of rehabilitation of return (gravity) piping in the section from FCT to CPS was implemented during operation in 2015 and 2016. Digging out of pipes, removal of the old insulation, cleaning and inspection of the pipes, repair of the surface and application of new waterproofing as needed. Repair of damaged piping was carried out by cold sleeves and build-on welding technology. The leaks that arose when the pipeline was uncovered were repaired with hot sleeves. At the same time cathode protection of the pipeline was applied during rehabilitation.
- Project of replacement of supply pipes in the section from CPS up to the concrete monolith and return pipes in the section from the concrete monolith to FCT was realized during the joint outage of units 3 and 4 of NPP EBO V2 in 2016. The Project included replacement of the original buried pipes with new pipes, building inspection shafts at the concrete monolith, where the new pipes connect to the original piping in the concrete monolith, connection of the piping in CPS and FCT, and replacement of selected shut-off valves in the turbine hall and DGS. Pipeline led above ground has thermal insulation and was fitted with self-regulating heating cable to protect it against freezing.

In NPP EMO, it was not necessary to replace any piping component based on the results of measurements of wall thickness of piping components.

In NPP MO3&4, the “zero” check measurements of ESW piping wall thickness was carried out to confirm compliance with the design data.

4.3 Assessment and conclusions of the regulator on ageing management of concealed pipework

Ageing management of concealed pipes is a part of AMP for the ESW piping [76] that is implemented by the licence holder. This guide is valid for the NPP EBO V2 and NPP EMO in operation. For the NPP EMO units 3&4 under construction, the AMP will become effective before their commissioning. The scope of AMP for ESW (corrosion monitoring, monitoring of the concrete monolith, wall thickness measurement, visual inspections) covers monitoring of all relevant degradation mechanisms identified based on operational experience, international recommendations and results of the ageing management program. Based on monitoring of the status of ESW pipes, the reconstruction and replacement of these pipes was realized at the NPP EBO V2.

After refuelling of each operating unit and before its start-up, the results of in-service inspections of classified equipment are assessed with the participation of ÚJD SR. Thereafter, the licence holder sends to the ÚJD SR reports from the preliminary assessment of the results and from evaluation of results of in-service inspections of classified equipment in a form of regular reports according to the ÚJD SR´s decision [3] and the ÚJD SR issues its opinion on these. In-service inspections of classified equipment are performed by the licence holder according to the programs of in-service inspections approved by the ÚJD SR. In the course of planned ÚJD SR inspections aimed at checking the implementation of in-service inspections and inspections after refuelling, the ÚJD SR inspectors by random selection check also the documentation prescribed by ageing management programs for concealed pipes. In addition to planned inspections as mentioned above, the ÚJD SR conducts also unplanned inspections in response to various events in the world. Ageing management is subject of inspections focused on periodic safety review.

In most cases, inspections were without negative findings. In case of a negative finding the ÚJD SR issues a Protocol to the licence holder containing corrective actions and their deadlines, and subsequently checks fulfilment of these corrective actions within the specified deadlines. No major deficiencies were identified during inspections that would require immediate corrective actions. The ability of systems, structures and components relevant for nuclear safety to fulfil their safety functions is secured.

5 Reactor pressure vessels

The ageing management program for the reactor pressure vessel (RPV) implemented by the licence holder in accordance with the guide [83]. This guide is valid for all nuclear units in Slovakia, i.e. the NPP EBO V2, NPP EMO in operation and NPP EMO units 3&4 under construction. AMP RPV [83] has been developed according to the national [5] and international guides and recommendations, and contains the basic attributes of an effective ageing management program [1].

The ageing management program for RPV has been applied from the start of operation of NPP EBO V2 and NPP EMO, and covers also the RPV installed at the power plant under construction, NPP MO3&4.

5.1 Description of ageing management programs for RPV

5.1.1 Scope of ageing management for RPV

The AMP for RPV focuses on monitoring of the main degradation mechanisms, which include radiation embrittlement, fatigue and corrosion damage. Monitoring the effect of stress corrosion and of the wear-out is conducted by non-destructive tests during pre-operational or operational inspections and maintenance interventions. Within the AMP for RPV the effect of thermal ageing on the primary circuit materials is also monitored through the project of thermal ageing monitoring.

The role of the ageing management program for RPV is the following:

- To ensure operability and reliability of RPV including nozzles, sealing joint and head during design lifetime, as well as planned long-term operation of nuclear units of SE, a. s.,
- To collect and provide information on use of RPV lifetime and prediction of its development.

The ageing management program for RPV covers the following basic areas:

- Basic information,
- Identification of the degradation mechanisms,
- Data collection and recording,
- Assessment of the current status of RPV,
- Corrective actions and feedback,
- Outputs from the ageing management program.

5.1.1.1 Basic information on RPV

This chapter contains information on the basic RPV parts, which are the focus of the ageing management program. These parts include:

- Body of the pressure vessel including the base material, welds and austenitic build-on welding,
- RPV head, including TK, NFM and CA nozzles,
- RPV inlet and outlet nozzles,
- Sealing joint of the main dividing plane of RPV.

5.1.1.1.1 Body of the RPV including the base material, welds and austenitic build-on welding

The reactor pressure vessel is a steel vertical cylindrical vessel welded from six forged rings and hemispherical bottom. The RPV body is terminated by flange ring, on which the main dividing plate is formed, with threaded holes for the main flange coupling stud bolts, two pairs of grooves for the nickel sealing rings and core barrel flange for core barrel mounting.

Under the flange ring there are two sections of nozzles, between which there is a separating ring located from the inside of the vessel. Below the bottom section nozzles there is a support ring, by which RPV is attached to the support frame of the concrete shaft. The RPV is connected to the other primary circuit equipment by means of piping of six circulating loops and piping of safety and emergency systems. All piping of the downstream piping systems is connected to the RPV by welded joints. Checking and measuring devices are also connected with welded joints. Detachable joint is the sealing joint between the upper block with the main dividing plate of RPV.

The RPV body (rings and the bottom) is made of chrome-molybdenum-vanadium steel, 15Ch2MFA type according to GOST, while part of the vessel surrounding the reactor core, is made of steel with very low concentration of accompanying impurities. The permissible concentrations are determined to provide the required radiation resistance of the vessel material. The individual RPV rings, including the bottom, are welded using Sv10ChMFT material. In order to ensure corrosion protection of the RPV, there is cladding from austenitic stainless steel on all inner surface. Reactor pressure vessels of WWER 440/V213, installed at NPP EBO V2, NPP EMO and NPP MO3&4, were manufactured on the basis of Soviet technical design in "Škoda – Jaderné strojírenství Plzeň".

5.1.1.1.2 Reactor pressure vessel head, including TK, NFM and CA nozzles

The RPV head is a forging of hemispherical shape with nozzles for the CA drive sleeves in the upper part, TK and NFM nozzles located on the circumference and the projection for sealing the main dividing plane at the circumference of the head body. The body of the RPV head is made of 18Ch2MFA material, the base material of the CA, NFM and TK nozzles is 22K. Internal clad on the reactor head is made of the same material as for the pressure vessel.

5.1.1.1.3 Inlet and outlet RPV nozzles

Nozzles on the RPV secure the connection with the other systems of the primary circuit and are located on the upper and bottom section as follows:

- Upper section or nozzles:
  - 6 nozzles on Dn500 for connection to the main circulation pipe
  - 2 nozzles on Dn250 for connection to the emergency systems (ECCS)
  - 1 I&C nozzle
- Bottom section of nozzles:
  - 6 nozzles on Dn500 for connection to the main circulation pipe
  - 2 nozzles on Dn250 for connection to the emergency systems (ECCS).

Flange extensions Dn500, ECCS nozzles and I&C nozzle are made of austenitic material 08Ch18N10T and are welded to the RPV body by a heterogeneous weld joint.

5.1.1.1.4 Sealing joint of the main dividing plane of the RPV

The function of the sealing joint of the main dividing plane is to provide tightness of the detachable joint between RPV and the head of the upper block and consists of:

- Stud bolts,
- Complete loose flange,
- Complete nickel gaskets,
- Upper and lower washers.

The loose flange is made of 25CH3MFA, the bolts are made of 38CHN3MFA.

The reactor pressure vessel scheme, including the RPV head are shown in Annex 6.

5.1.2 Assessment of RPV ageing

Ageing management program focuses on the following RPV areas:
- Pressure vessel body including base material, welds and austenitic build-on welding,
- Pressure vessel head, including TK, NFM and CA nozzles,
- Inlet and outlet nozzles on RPV,
- Sealing joint of the main dividing plane of RPV.

5.1.2.1 Identification of ageing mechanisms

The reactor pressure vessel operates under conditions that cause a change in the properties of the vessel materials by the action of multiple degradation mechanisms of varying intensity and in different parts of RPV. Identification of degradation mechanisms is based on operational experience and national and international recommendations (IGALL).

The degradation mechanisms affecting RPV include:
- Radiation embrittlement,
- Fatigue damage,
- Thermal ageing,
- Stress corrosion,
- Corrosion,
- Wear-out,
- Loss of pre-load of bolt connections.

There are degradation mechanisms demonstrated in different parts of RPV as shown in Table 5-1.

Table 5-1 Degradation mechanisms in individual parts of RPV

	Radiation embrittlement	**Fatigue damage**	**Thermal ageing**	**Stress corrosion**	**Corrosion**	**Wear**
Pressure vessel body including base material, welds and clad	X	X	X	-	X	-
Pressure vessel head, including TK, NFM and CA nozzles	-	-	X	X	X	-
Inlet and outlet nozzles of RPV	-	X	X	-	X	-
Sealing joint of the main dividing plane of RPV	-	X	-	X	X	X

5.1.2.2 Description of degradation mechanisms

5.1.2.2.1 Radiation embrittlement

By irradiating the material with fast neutrons, there are significant changes in mechanical properties of the construction materials. Due to irradiation, the yield strength values and tensile strength significantly increase, the hardness also increases and the fracture and notch toughness decrease. Radiation embrittlement is mainly influenced by the density of the neutron flux, the neutron energy spectrum, the irradiation temperature, chemical composition and microstructure of materials around the reactor core (the weld and the base material in the surroundings).

Radiation embrittlement is considered to be the prevailing degradation mechanism affecting the reactor pressure vessel, with high significance, which is given special attention.

5.1.2.2.2 Fatigue damage

Fatigue damage is caused by cyclic load on the components, so it is often referred to as accumulation of fatigue damage. Fatigue damage occurs in two stages. In the first stage, the accumulation irreversible local plastic deformations lead to initiation of cracks (micro-cracks). In the second stage, cyclic growth from the initialized crack follows to reach critical dimensions, where the damage process can be completed by the final fracture of the structure. In order to assess and predict the process of damage resulting from material fatigue, there are physically supported and mathematically sophisticated procedures codified in the national normative standards, such as ASME and PNAE-G. For the evaluation of fatigue damage it is necessary to know the properties of the materials and the load history (pressure, temperature and lapsed unit modes). The most stressed joint in terms of fatigue damage are the nozzles at the inlet and the outlet from the reactor pressure vessel.

5.1.2.2.3 Thermal ageing

One of the degradation mechanisms of the primary circuit materials is thermal ageing. This ageing mechanism is observed for components that are operated for a long time at temperatures 280 to 300°C. The consequence of this process is the gradual negative change in the mechanical properties of the material due to diffusion processes supported by the operating temperature of the material.

5.1.2.2.4 Corrosion

Corrosion is a type of material degradation that is caused by its chemical or electrochemical reaction to the surrounding environment. The effects of corrosion are related to the existence of a corrosive environment consisting of metallic material and corrosive medium (environment). It manifests itself with changes in the structure and shape of the material. It is characterized by a loss of material and deterioration of its mechanical properties. One form of corrosion is attack due to contamination by boric acid (boric acid corrosion). Corrosion damage is primarily applied to the outer surfaces of reactor pressure vessel, reactor head and other structural parts that are not protected by austenitic build-up welding.

5.1.2.2.5 Stress corrosion

A special case of corrosion damage is corrosion cracking (stress corrosion cracking), which is characterized by quasi-brittle failure without detectable corrosion products. Its formation is conditioned by simultaneous action of three factors i.e. sufficiently high pulling stress, aggressive environment and sensitive material.

Corrosion cracking can be divided into three stages:

- The first stage ends with the initiation of micro-defect,
- The second stage is characterized by a stable defect growth,
- In the third stage there is a rapid unstable growth of macro-defect.

The final stage always occurs after exceeding certain threshold value of stress intensity coefficient K_{ISCC}. The course itself, the duration of individual stages, the rate of corrosion growth and the threshold value of K_{ISCC} are influenced by local properties of the material and the acting media (conductivity, corrosion potential, and chemical composition – especially oxygen content).

5.1.2.2.6 Wear-out

The wear-out is characterized by loss of material due to contact between two mutually moving surfaces of the components. It results in a continuous formation of plastically deformed exposed micro-surfaces. If a chemically active environment is present, chemical reactions may occur in certain areas of this excited surface. Subsequently, a surface oxide layer can be formed, which reduces the tendency to adhesion, reduces wear-out, and the friction factor. It can also lead to chemical reactions that accelerate the wear-out. Under certain conditions, the decomposition of surface corrosion oxide layer results in particles which, if they cannot be removed from the contact area, accelerate abrasive wear-out.

5.1.2.2.7 Loss of pre-load of bolt joints

This degradation mechanism is applied to bolt joints of the main dividing plane and bolts of CA drive nozzles. The loss of pre-load is a result of relaxation of tensions from increased temperature or due to vibrations, when loosening of bolt connections occur. The criteria for assessing the admissibility of loss of pre-load are set in the relevant instructions for the in-service inspections and are monitored each time the bolt connections are dismantled once a year.

5.1.3 Monitoring, testing, sampling and inspection activities for RPV

Assessment of the current status of the RPV is performed by monitoring the degree of degradation as a result of degradation mechanisms. The equipment status indicator is a defined characteristics that can be monitored or measured, and is used to estimate or directly indicate the current or future capability of the equipment or its part to fulfil its function within acceptable criteria.

To evaluate the degree of degradation of individual parts of RPV, data from:

- Assessment of radiation damage,
- Assessment of fatigue damage,
- Monitoring of thermal ageing,
- In-service inspections,
- Maintenance,
- Expert opinions and analyses are being utilized.

The following chapters describe the methods of monitoring of degradation mechanisms for individual assessed nodes of reactor pressure vessel, by means of surveillance specimen programs, monitoring, testing, sampling, in-service inspection programs, or other inspection activities for RPV.

5.1.3.1 The RPV body including the base material, build-up welding and weld

5.1.3.1.1 Assessment of radiation damage

Monitoring of fast neutron fluency

At NPP EBO V2 and NPP EMO the fast neutron fluency is continuously measured on the outer wall of RPV and is evaluated each time after the respective fuel campaign. The neutron fluency values are then used to calculate radiation damage of reactor pressure vessel materials in the reactor core, and that is for the base material and for the weld metal by determining the value of the critical temperature of brittleness T_{KF}.

Radiation damage is defined by change in the brittle facture temperature. To calculate the shift in the critical brittleness temperature the inputs used are values of source brittleness temperature of the material before irradiation (T_{K0}), the coefficient of radiation embrittlement (A_f) and fast neutron fluency (with the energy of 0.5 MeV) measured after the pressure vessel.

Source temperature T_{k0} is defined in the passport of reactor pressure vessel. Coefficient of radiation embrittlement A_f is experimentally obtained based on changes in the mechanical properties of RPV material within the surveillance specimen program of RPV.

Surveillance Specimen Program of RPV materials

Part of delivery of WWER 440/V213 reactors installed at NPP EBO V2 was a standard surveillance specimen program of RPV, containing original materials of the base and weld metal from the production. This program ran from commissioning the units in 1984/1985 (NPP EBO V2) during a period of 10 fuel campaigns. Upon completion of this program, a supplementary program of RPV surveillance specimens was applied NPP EBO V2, whose irradiation strings were loaded to the original irradiation channels of the reactor from 1995 to 1996. The program proceeded by gradual collection of irradiated samples during 10 fuel campaigns.

At NPP EMO, an upgraded program of surveillance specimens is introduced that has been running since the commissioning of units in 1998/ 2000 with a period of irradiation of samples during 10 campaigns.

At present, at NPP EBO V2 and NPP EMO, an improved surveillance specimen program is applied. Within this surveillance program, in the period 2008 to 2012 at NPP EBO V2, and from 2011 to 2013 at NPP EMO, irradiation chains were inserted in the channels with irradiation specimens. The program for one nuclear unit consists of six irradiation sets, which are taken out of the reactor step by step, according to the defined timetable.

Within the scope of surveillance specimen program, the maximum temperature achieved in individual irradiation capsules. These measurements are carried out by means of eutectic alloys, which are characterized by their melting at reaching a certain temperature.

An improved surveillance specimen program includes samples from the following materials:

- Weld metal,
- Austenitic build-on welding,
- HAZ near the weld in the area of reactor core,
- HAZ under the build-on welding,
- Austenitic material 08Ch18N10T (representing reactor internals material).

For the NPP MO3&4, there is a surveillance specimen program for RPV materials, which in total is made of 8 irradiation chains with irradiation capsules, containing besides the RPV materials also materials of reactor internals. These will be evaluated after exposure of 10 and 20 campaigns.

The evaluation of mechanical properties of materials, included in the surveillance program is performed by the following destructive tests: static fracture toughness test, the Charpy impact test and SPT test:

- Static fracture toughness test – for testing, the yield method on test elements with a cyclic crack is used. The yield method includes gradual loading of the test piece with partial periodic relaxation within the whole temperature range. From the test results, experimental values of the static fracture toughness are calculated. From the regression curves of temperature dependence of the fracture toughness, the values of transition temperature TT_{100LH} are deducted at a value of static fracture toughness of 100 $MPa.m^{1/2}$.
- The Charpy impact test – (bending shock test) the test specimen with notch is used.
  The main criterion is:
  - The energy criterion (i.e. temperature, at which the value of impact energy is 50 J/cm^2)
    The secondary criteria are:
  - The transverse extension criterion (T_k is determined for the value ∂ = 0.9 mm)
  - The fracture toughness criterion (T_k corresponds to 50 % share of ductile fracture measured on the Charpy specimen fracture area).
    Experimentally obtained values of temperature dependence of notch toughness are translated by the least square method to regression curves for BM and WM. From the regression curves, the values of the transition temperature to brittle failure are deducted at the value of notch toughness of 50 J/cm^2.
- SPT test – the test represents a progressive method for determining mechanical properties of materials. The basic advantage of this test is the size of the test element (8 mm in diameter and 0.5 mm thick) with a need of relatively small volume of material, which allows to evaluate the properties of the heat affected zone in the vicinity of a weld joint, the heat affected zone under the austenitic weld-on or the austenitic weld-on RPV. These types of materials are the subject of an improved surveillance sampling program, which is currently applied at NPP EBO V2 and NPP EMO.
  The transition temperature of the brittle and tough fracture obtained from the temperature dependence of consumed SPT energy is comparable with the results obtained with Charpy-V test. The fracture behaviour of materials used on nuclear power facilities can be assessed based on fracture energy from SPT test. Obtained SPT test results are used to determine the yield strength and tensile strength, the transition temperature between brittle and notch fracture, or to assess the fracture properties of the test material.
  For an overall assessment of the state of material and its properties, tests are performed at different test temperatures [84].

Acceptability criteria

Maximum admissible values of brittle temperature T_{ka} for irradiated material of welded joint in the area of reactor core were set based on PTS analyses for RPV at NPP EBO V2 and NPP EMO. This analysis was elaborated according to VERLIFE [32] project, containing recommended procedures for lifecycle analyses of NPP equipment of WWER and the IAEA guide [85] for PTS analyses of NPP with WWER. As the worst case scenario, the LOCA 25 event on the hot branch of the main circulation node No. 1 was considered.

The maximum permissible critical brittle temperature T_{ka} for the weld in the area of reactor core is the following:

- Units 3&4 of NPP EBO V2 99.4°C,
- Units 1&2 of NPP EMO 107.3°C.

5.1.3.1.2 Assessment of fatigue damage of RPV

On the basis of pre-operational analysis, an area with significant accumulation of fatigue damage was identified on the RPV body on the cylindrical part in the area of weld in the reactor core. When calculating fatigue damage, the following information is taken into account:

- Actual measured operating data on temperature and pressure in the PC,
- Operational history of the unit with precise specification (date and time) of operating modes according to the L&C,
- Number of pressure tests of PC and SC,
- Results of NDT inspections of RPV.

The assessment of fatigue damage is done always after the end of the relevant fuel campaign. As a result of fatigue life calculation analysis the accumulation of damage is expressed as percentage for the period from start of operation until the current campaign.

Within the framework of operational safety indicators, the following assessment criteria are currently adopted for assessment of fatigue life use of the most stressed RPV nodes:

- Strategic objective 1.0 % per campaign,
- Planned value 1.2 % per campaign,
- Limit of acceptability 1.5 % per campaign.

5.1.3.1.3 Monitoring of thermal ageing

Monitoring of thermal ageing of materials in SE, a. s. is ensured through a project, within which RPV material samples are placed into operating environment for a long-term heat exposure. Part of the project is sampling of material realistically under thermal load through surface sampling of the body of reactor pressure vessel. The project output is quantification of changes in material properties due to thermal ageing realized by evaluation of mechanical tests on SPT samples – determining elongation at break, yield strength and strength limits, measuring micro-hardness and micro-structural analysis of samples.

Within the TAM project, at RPV of NPP EBO V2 and NPP EMO, the following activities are performed:

- Evaluation of RPV materials in “0-state”,
- Exposure of RPV material via a sample carrier located on the main circulation piping,
- Sampling of RPV surface material.

Criteria for assessment of thermal ageing
Evaluation of RPV material in “0-state” is performed as one-time establishment of initial material properties. Sampling of surface RPV material is performed as required. Exposed RPV material located on the main circulation pipe is evaluated according to the defined timetable of the TAM project. Sufficient amount of archival material is a condition for this assessment.

Criteria for the assessment of exposed materials are defined by the relevant norms for individual tests.

5.1.3.1.4 Corrosion

Due to the fact that corrosion processes cannot be accurately predicted on the basis of current knowledge, for their monitoring the main emphasis is placed on the combination of periodic NDT in-service inspections with subsequent expert assessment of the type of corrosion process.

5.1.3.1.5 In-service inspections

The NDT in-service inspections of RPV in SE a. s. are performed in a regular 8 year cycle. Inspections are carried out from outer and inner surface, with the inspection from the inner surface being carried out after a complete export of fuel during RGO. Inspection from the outer surface is carried out with a 4 year delay against the inspection from inside. Inspections from inside and from outside are performed according to uniform technical specification, while respecting the differences in accessibility for inspection of individual check points from outer and inner surface.

The following points of the pressure vessel body are subject to NDT inspections, including the base material, welds and austenitic build-up welding,

- All circumferential welds, including base material in thermally affected area,
- Selected areas of base material on each RPV ring,
- Selected part of the elliptic bottom.

Checks from the inner surface of RPV are performed by using the following NDT methods:

- Ultrasonic test of welds, base material and build-up welding,
- Eddy current test of the build-up welding surface,
- Indirect visual test of inner surface.

Checks from the outer surface of RPV are performed by using the following NDT methods:

- Ultrasonic test of welds, base material and build-up welding,
- Indirect visual test of outer surface.

All checks are mechanized, automated and use modern testing devices and control systems. The ultrasonic testing of RPV body tests the inner and outer surface of the material over the entire thickness of the RPV wall. The ultrasonic test and the eddy current test of all check points are qualified according to the European methodology, based on the requirements of UJD SR. Qualification criteria for the ultrasonic material test near the welding interface of the build-up welding are determined by VERLIFE methodology.

The program of operational NDT and technical inspections for NPP EBO V2 and NPP EMO is defined in the relevant operating documents [41], [42], [38], [39].

5.1.3.2 Reactor pressure vessel head, including TK, NFM and CA nozzles

5.1.3.2.1 Monitoring of stress corrosion

Stress corrosion is a special case of corrosive damage, resulting in a quasi-brittle failure of material without detectable corrosion products. Important construction locations are characterized by the presence of three factors:

- Sufficiently high pulling stress,
- Aggressive environment,
- Sensitive material.

5.1.3.2.2 Corrosion

Due to the fact that corrosion processes cannot be accurately predicted on the basis of current knowledge, for their monitoring the main emphasis is placed on the combination of periodic NDT in-service inspections with subsequent expert assessment of the type of corrosion process.

5.1.3.2.3 In-service inspections

Within the NDT checks the following inspections are performed on the pressure vessel head, including TK, NFM and CA nozzles in the following areas:

- Sealing surfaces of the head and sealing surfaces of TK, NFM and CA nozzles,
- Build-up weld on the inner surface,
- Girth weld of the head,
- Flange,
- Internal and external welds of TK, NFM and CA nozzles with the head.

NDT and technical inspections are performed at prescribed intervals and within the scope of relevant operating documents [41], [42], [38] and [39].

5.1.3.3 Inlet and outlet RPV nozzles

5.1.3.3.1 Assessment of fatigue damage of RPV nozzles

Based on the pre-operational analysis, an area with significant accumulation of fatigue damage on the inlet and outlet RPV nozzles was identified. For calculation of the fatigue damage, the following documents are required:

- Actual measured operating data - temperature and pressure in PC,
- History of unit operation with exact specification (date and time) of lapsed modes according to LCO,
- Number of successful and failed pressure tests of the PC,
- Results of NDT RPV inspections.

The assessment of fatigue damage is done always after the end of the relevant fuel campaign. As a result of fatigue life calculations, accumulation of damage is expressed as percentage of the period from the start of operation up to the current campaign.

Within the system of operational safety indicators, the following assessment criteria are currently adopted for assessing use of fatigue life of the most stressed RPV nodes:

- Strategic objective – 1.0 % per campaign,
- Planned value – 1.2 % per campaign,
- Acceptability limit – 1.5 % per campaign.

5.1.3.3.2 Corrosion

Due to the fact that corrosion processes cannot be accurately predicted on the basis of current knowledge, for their monitoring the main emphasis is placed on the combination of periodic NDT in-service inspections with subsequent expert assessment of the type of corrosion process.

5.1.3.3.3 In-service inspections

Within the in-service inspections, the following construction areas are checked at the RPV inlet and outlet nozzles:

- Welds and base material of Dn500 and Dn250 nozzles,
- Internal radial passage of Dn500 nozzles.

NDT and technical inspections are performed at prescribed intervals and within the scope according to the relevant operating documents [41], [42], [38], [39].

5.1.3.4 Sealing joint of the RPV main dividing plane

5.1.3.4.1 Assessment of fatigue damage

On the basis of the pre-operational analysis, an area with significant accumulation of fatigue damage on the sealing joint of the RPV main dividing plane was identified. For calculation of the fatigue damage, the following documents are required:

- Actual measured operating data - temperature and pressure in PC,
- History of unit operation with exact specification (date and time) of lapsed modes according to LCO,
- Number of successful and failed pressure tests of the PC,
- Results of NDT checks.

The assessment of fatigue damage is done always after the end of the relevant fuel campaign. As a result of fatigue life calculations, accumulation of damage is expressed as percentage of the period from the start of operation up to the current campaign.

Within the system of operational safety indicators, the following assessment criteria are currently adopted for assessing use of fatigue life of the most stressed RPV nodes:

- Strategic objective – 1.0 % per campaign,
- Planned value – 1.2 % per campaign,
- Acceptability limit – 1.5 % per campaign.

5.1.3.4.2 Stress corrosion

Stress corrosion is a special case of corrosive damage, resulting in a quasi-brittle failure of material without detectable corrosion products. Important construction locations are characterized by the presence of three factors:

- Sufficiently high pulling stress,
- Aggressive environment,
- Sensitive material.

Locations sensitive to this type of failure are detected through in-service inspections described in point 5.1.3.4.5.

5.1.3.4.3 Corrosion

Due to the fact that corrosion processes cannot be accurately predicted on the basis of current knowledge, for their monitoring the main emphasis is placed on the combination of periodic NDT in-service inspections with subsequent expert assessment of the type of corrosion process.

5.1.3.4.4 Wear-out

The wear-out is characterized by loss of material due to contact between two mutually moving surfaces of the components. Locations that are sensitive to this type of failure are detected through operational NDT checks.

5.1.3.4.5 In-service inspections

Within the framework of in-service inspections, on the main dividing plane joint, the inspections of the following construction areas are checked:

- Weld-on of sealing surfaces and grooves of the main sealing plate,
- Connecting material.

NDT and technical inspections are performed at prescribed intervals and within the scope according to the relevant operating documentation [41], [42], [38] and [39].

5.1.3.5 Maintenance interventions

The type and number of maintenance interventions on the equipment is an important indicator for assessment of the equipment status. In terms of equipment lifetime, those interventions are important, when a repair was performed on the equipment or replacement of its part due to inadmissible indications. These include, in particular: grinding, welding, build-up welding, mechanical machining and replacement of equipment or its part.

5.1.4 Preventive and corrective actions for the RPV

In preparing a corrective action, the root cause of excessive ageing is identified for the monitored point on RPV, and measures are proposed to eliminate the causes of degradation.

To this end, a working group is set up to analyse the causes of increased effect of degradation mechanism, and to propose effective measures to reduce the trend of degradation mechanism effects (e.g. by proposing a change in SSC material, operating modes, etc.).

The working group summarizes all available information on the state, operation and maintenance of the component, and also finds whether there were any design changes made on SSCs that could affect the operation of SSC.

The working group proposes work procedures to better understand the relationship between the SSC operation and the effect of the degradation mechanism. If necessary, additional analyses are performed, to help better understand the effects of the degradation mechanism and the circumstances causing it.

Based on the assessment of the degradation mechanism after several fuel campaigns, the working group will evaluate the effectiveness of measures to mitigate the effects of the degradation mechanism.

If the status indicator was found to achieve the limit value represented by the acceptability criteria, the working group proposes solutions in a form of repair of part of SSC, replacing the entire SSC with a new one or based on analysis proposes life extension of SSC under the condition of fulfilment of adopted and approved measures for a certain period of operation. In such cases, these proposals are supported by analyses and submitted to the regulatory authorities for approval well in advance.

5.2 Experience of the licence holder with the application of AMP for the RPV

The scope of activities within the AMP for RPV covers monitoring of all relevant degradation mechanisms identified on the basis of operational experience, international recommendations and outcomes of the ageing management program.

Based on the results of the RPV ageing management program it can be stated that none of the values of the monitored degradation mechanisms exceed the acceptability criteria, and on the basis of the forecast with the current mode of operation during the long-term operation, will not reach their limit values.

As a significant progress in monitoring degradation of RPV materials can be considered extension of surveillance programs with new materials located in the RPV core that have not yet been assessed. This is a material of heat-affected zone (HAZ) near the weld, HAZ under the build-up welding and the austenitic build-up welding itself. Since in these cases the material is from narrow structural areas, the SPT method is used for their assessment, which is a progressive method for monitoring changes in the mechanical properties of the material.

Significant progress has been achieved in assessing the RPV fatigue damage. New computational models have been developed with a new sufficiently fine sub-division of assessed nodes of the RPV body into final elements, which ensures reliable distribution of deformations resulting from its loading and thus also recalculated stress.

For the power uprate of nuclear units in NPP EBO V2 and NPP EMO, calculations were made for the fluency of fast neutrons falling on the RPV wall at an increase reactor power of 107% for a long-term operation.

Another extension in the AMP of RPV was introduction of thermal ageing monitoring program of RPV materials, which included collection of surface samples from the area of the weld and the RPV base material, and samples of RPV materials were placed into operating environment for long-term heat exposure.

Degradation mechanisms, such as radiation embrittlement, fatigue damage and thermal ageing require specific activities based on sampling, fluency monitoring, expert analyses and assessments. Other degradation mechanisms (corrosion, stress corrosion and wear-out) are monitored in the framework of NDT checks.

5.3 Assessment and conclusions by the regulator on the RPV ageing management

The ageing management program for the reactor pressure vessel has been implemented by the licence holder in compliance with the guide [83]. This guide is valid for all nuclear units in SR, i.e. NPP EBO V2, NPP EMO in operation and NPP units MO3&4 under construction. The scope of activities within AMP of RPV (surveillance specimen program, monitoring fluency, fatigue damage assessment, in-service inspections) covers monitoring of all relevant degradation mechanisms identified on the basis of operational experience, international recommendations and results of the ageing management program. The surveillance specimen program was extended to include new materials located in the reactor core, and covers operational conditions under increased unit power, and use of new type of nuclear fuel for a period of 60 years. The OSART 2010 mission at NPP EBO V2 identified this program as “Good Practice“.

After refuelling of each operating unit and before its start-up, the results of in-service inspections of classified equipment are assessed with the participation of ÚJD SR. Thereafter the licence holder sends reports to the ÚJD SR from the preliminary assessment of results, and from evaluation of results of in-service inspections of classified equipment, in a form of regular reports according to the ÚJD SR´s Decision [3] and the ÚJD SR issues its opinion on these. In-service inspections of classified equipment are carried out by the licence holder according to the programs of in-service inspections approved by the ÚJD SR. In the course of planned ÚJD SR inspections aimed at checking the implementation of in-service inspections of classified equipment and inspections after refuelling, the ÚJD SR inspectors by random selection also check the documentation prescribed by the ageing management program for the RPV. In addition to the planned inspections, the ÚJD SR conducts also unplanned inspections in response to different events worldwide. Ageing management is also subject of inspections for periodic safety review.

In addition to the evaluation of results of in-service inspections of classified equipment, the licence holder provides also an information on use of lifetime of classified components, and thus also RPV. The licence holder submits a report on the state of use of RPV lifetime in a form of regular reports according to the ÚJD SR´s Decision [3] and the ÚJD SR issues its opinion.

In most cases, inspections were without negative findings. In case of a negative finding, the ÚJD SR issues a Protocol to the licence holder containing corrective actions and their deadlines, and subsequently checks fulfilment of such corrective actions within the specified deadlines. The inspections did not identify any major deficiencies that would require immediate corrective action. Capability of the systems, structures and components relevant for nuclear safety to fulfil their safety functions is ensured.

6 Calandria/ pressure tubes (CANDU)

Not applicable for Slovakia. Calandria/ pressure tubes are not used in nuclear installations in Slovakia.

7 Concrete containment structures

The ageing management program for the containment of nuclear power plants is implemented by the licence holder and its implementation is in compliance with the guide [86]. This guide is valid for the nuclear units in operation in SR, NPP EBO V2 and NPP EMO. For the units of NPP MO3&4 in the phase of construction the AMP for the containment will be put into effect before their start-up. The Program has been developed in accordance with the national [5] and international guidelines and recommendations, and contains the basic attributes of effective AMP [1].

7.1 Description of ageing management programs for concrete containment structures

7.1.1 Scope of ageing management for concrete containment structures

The WWER 440/V213 nuclear power plant, both from structural and technological aspect, consists of two separately operated units 1&2, situated in the building of the main generating dual-unit. The reactor building is divided into containment and non-containment parts.

The containment of the WWER 440/V213 NPP represents the fourth physical barrier to the release of radioactive substances into the environment. It consists of the following main parts:

- Reactor shaft,
- Fuel exchange pool,
- Steam generators box,
- Accident localization shaft (vacuum-bubble condenser),
- Ventilation centre.

Containment fulfils the following basic functions:

- To prevent the release of radioactive materials outside the containment space, above the specified level, i.e. ensuring the required tightness and biological protection under normal operating conditions, as well as at the maximum design accident,
- To manage the effects of increased pressure and temperature inside the containment space as a result of the reactor cooling system accident,
- To protect the reactor cooling system and the plant equipment from external influences from outside and inside of the containment. Containment must retain its design function during the effects of natural hazards, such as: earthquakes, wind, rain, extreme temperatures and human activities (air and car transport, fires, etc.).

The containment is sub-atmospheric, with the possibility to reduce the pressure in case of design accident with a vacuum-bubble condenser and spray system. It is designed to 150 kPa atmospheric overpressure and 20 kPa atmospheric negative pressure to adequately cover the design load at the design basis accident, and meets the current IAEA requirements for the third physical barrier of the defence-in-depth concept. The boundary of the containment is formed by reinforced concrete walls with hermetic steel liner that separate all containment spaces from the surroundings, and other structural, technological and electrical elements of the containment that ensure its tightness. These are, in particular, hermetic doors, covers, penetrations (pipe, impulse, electrical, shaft), protective reactor shaft lid, refuelling pool stop-log, fast acting valves and HVAC elements. The containment steel liner binds all its parts and thus guarantees the gas-tightness of reinforced concrete walls.

On the pipelines of technological systems that cross the containment boundary, there are fast-acting separating valves installed on both sides, and they are designed to quickly isolate the systems inside the containment in case of an accident.

The capability of the vacuum-bubble condenser to fulfil its design function during the design basis accident was demonstrated by its qualification [87].

During operation of the reactor unit, the containment has a negative pressure of min. 50 Pa. Operation of the containment with a constant negative pressure enables continuous monitoring of its tightness, which is a significant safety aspect of this type of containment. The integrity of the containment is verified by periodic tightness tests at internal overpressure.

In case of abnormal operating conditions the containment provides protection against the release of radioactive substances to the surroundings of the NPP and to the environment.

The scheme of WWER 440/V213 containment is shown in Annex 7.

The basis for determining the extent of ageing management for concrete containment structures were outputs from research projects [88], [89], implemented in the period from 2002 to 2008 in co-operation with expert organizations. Within the given projects, individual elements of containment building structures were analysed in terms of possible degradation mechanisms within the design, as well as the long-term operation. To assess the influence of temperature and radiation load on the change of concrete properties, samples were taken from the wall of the reactor shaft at NPP EBO V2. A comprehensive analysis of these samples showed that the influence of then 20 years of operation on the change in properties of the concrete is insignificant, and the concrete fully meets design requirements.

On the basis of the outputs and recommendations from the above projects, an ageing management program was developed for the main generating unit, including concrete structures of the containment.

The AMP for the containment is designed in a way to achieve:

- Increasing the safety level by early prediction of the state of building structures of the containment in individual NPPs in the SR,
- Assessment of the actual state of use service life and its trends,
- Conceptual influence of the conditions of use of building structures of containment,
- Best knowledge of the state of building structures of containment for long-term operation and life extension.

The goal of AMP is to ensure reliability and safety of monitored building structures of the containment through the knowledge of:

- Initial condition,
- Materials and material conditions,
- Ageing mechanisms,
- Places of degradation,
- Degradation degree indicators,
- Consequences of degradation by ageing and the resulting failures under normal and abnormal operating conditions.

The containment AMP includes building structures and their elements relevant for the integrity of the containment. These are the following building structures:

- Supporting reinforce concrete structure,
- Steel structure,
- Steel liner (hermetic, non-hermetic),
- Elements of the technological equipment built-in the building structure (containment penetrations, doors, covers).

7.1.2 Assessment of containment concrete structures ageing

Identification of individual degradation mechanisms, their monitoring and assessment forms the basic part of a comprehensive containment AMP.

From the point of view of possible degradation of the building structure, the critical locations are determined on the basis of assessment of the characteristics of the environment in the rooms under normal operation, as well as during emergency conditions. For the given environment consideration will be given to the possibility of occurrence of individual degradation mechanisms.

Based on the impact on nuclear safety and stressors (temperature, humidity, radiation, mechanical stress, chemical effects, weather conditions) the rooms were identified or parts of building structure, in which the degradation mechanisms have the greatest impact on the degradation of individual building structures of the NPP containment.

The elements of the containment structures that exhibit degradation mechanisms are the following:

- Concrete,
- Steel reinforcement of concrete,
- Steel liner,
- Anchoring elements,
- Load-bearing and non-load bearing steel structures,
- Penetrations, covers, manholes, doors,
- Seals and sealing elements,
- Protective coating.

Identification and quantification of operation of degradation mechanisms on the individual elements of building structures of the containment is carried out by non-destructive and destructive tests. Due to the fact that destructive tests can be done mainly on samples taken outside the NPP site, the main emphasis in the implementation of inspections is on non-destructive tests. Destructive tests are used in particular if with non-destructive tests the part of building structure cannot be assessed within the required scope.

When examining individual building structures, attention is focused on identification and quantification of the following chemical, physical and mechanical events:

- Concrete and reinforced concrete,
  - Leaching and blooming
  - Effect of sulphates
  - Effect of acids and alkali
  - Reactions of alkali metals with filler
  - Carbonising
  - Crystallization of chlorides and other salts

- Corrosion of steel reinforcement
    - Effect of freezing and melting
    - Abrasion, erosion
    - Increased temperature, thermal cycling
    - Influence of radiation
    - Fatigue, vibration
    - Settlement
- Steel structure and steel liner,
    - Corrosion
    - Increased temperature, humidity and thermal cycling
    - Fatigue, vibration
    - Influence of radiation

In identifying the effect of degradation on individual building structures it is important to determine the cause of the degradation due to the fact that the various degradation mechanisms exhibit the same form of degradation.

From the view of comprehensive evaluation of the actual state of individual building structures, the licence holder in cooperation with the supplier has developed and implemented a methodology for assessing used lifetime of the containment. The methodology, based on weighting factor established for each assessed degradation mechanism, makes it possible to evaluate the current state of containment structures with respect to design assumptions or to compare the status on individual units.

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

Since the individual units of NPP EBO V2 and NPP EMO have been put into operation, the tightness tests and test of the strength of containment are periodically performed. These tests are aimed at verifying its safety function – to prevent release of radioactive substances outside the containment space. Since the start-up of units, there are regular geodetic measurements made and evaluation of settlement of the main generating unit building.

With regard to the introduction of a comprehensive AMP for building structures of the containment, the original monitoring program was extended by new procedures and areas of inspections of individual components of building structures of the containment.

Status monitoring is performed according to AMP periodically on each operating unit according to the defined plan.

The inspection plan is structured according to three basic types of survey:

- Basic construction and technical survey,
    - Visual inspection of the building, its structures and the closest surroundings
    - Mapping of detected faults
- Comprehensive construction and technical survey,
    - Visual inspection of the building, its structures and the closest surroundings
    - Non-destructive testing of selected parts and elements of individual building structures - contact methods, small sampling of material
    - Evaluation of cracks and cracks map update
    - Realization: once every 4 years during the GO of the relevant unit

- Special construction and technical survey,
  - Predominantly destructive methods requiring sampling of tested items and their laboratory evaluation
  - Implemented in case of requirements to determine properties that cannot be evaluated by non-destructive methods.

The evaluation of the results of monitoring is carried out according to the acceptance criteria for the acceptability of the degree of degradation and its further development over time. For assessing the degradation state of reinforced concrete structures, the criteria defined in the national standard STN 731201 or Eurocode 2, are used.

Within the framework of individual types of surveys of building structures of containment, the inspections are focused on the following:

- Basic construction and technical survey/area of inspection,
  - The integrity of coatings of building structures – concrete or steel
  - Delamination of concrete or paint
  - Joints, gaps, connections (loose joining material)
  - Deformations of concrete or steel structures
- Comprehensive construction and technical survey/ area of inspection,
  - Wet places, cavities, crystallization salts
  - Concrete strength, compactness, absorption, carbonization and moisture
  - Cracks (crack activity)
  - Diagnostics of reinforcement and failures of building structures using geo-radar
  - Thickness of the liner and coatings
- Special construction and technical survey/area of inspection,
  - Laboratory tests of strength and concrete microstructure
  - Quality analysis of concrete
  - Laboratory tests of samples from steel structures and anchor elements.

One of the outputs of a comprehensive construction and technical survey of concrete structures of the containment carried out within the framework of AMP is the identification of the cracks map, their schematization and measurement. Permissible cracks are assessed according to Eurocode 2. Based on the data acquired about cracks (type, location, length, width, movement) in the concrete structures of the containment, the so called cracks map is developed [90].

Monitoring of the environment

Taking into account the fact that temperature and radiation may have a significant impact on the possible degradation of properties of individual containment structures, the licence holder has introduced a program for monitoring the environment in the designated locations of the containment. The purpose of monitoring is to verify the consistency between the design values of temperature and radiation in the exposed areas of containment and the real condition during operation.

Care program for hermetic covers and doors

The licence holder has developed and implemented a program for hermetic covers, closures, doors that are important for ensuring the required containment tightness. At specified intervals, the proper functioning of cover seals and door seals is checked, and the individual components of the gaskets are replaced. Properly applied care for hermetic covers and doors results in satisfactory result of the containment tightness test [91].

7.1.3.1 Operational tests of the containment required by the legislation

Integral strength test
The integral strength test is performed in order to assess the limit condition of ÚJD SR 3.6.1.2, which reads as follows: The strength of the outer containment boundary at an overpressure of 150 kPa is sufficient if no destructive or deformation changes are formed, caused by overpressure and maximum bending of walls on the Accident Localization Shaft (ALS) does not exceed 10 mm. The integral strength test is performed at least once in 10 years.

Periodic integral leakage test
The periodic integral test determines the percentage of air leakage from the containment during the 24 hour overpressure tightness test. It is performed once every 2 years [92], [93].
Up to now, the containment tightness and strength tests performed on operated units in Slovakia, did not indicate any problem with the tightness and integrity of the containment. Despite that, work is constantly being carried out to improve the containment tightness.

7.1.4 Preventive and corrective actions for concrete structures of the containment

The proposal of preventive and corrective actions is based on the results of assessment of individual ageing indicators or the impetus is gained experience from other operators. When corrective action is developed, identification of the root cause of the excessive ageing of the monitored place of the building structure is performed, and thereafter measures are proposed to eliminate the identified causes of degradation. Preventive and corrective actions are developed and approved by a working group, the composition and scope of powers of which is defined in the internal documentation of the licence holder.

Interventions into building structures of the containment may not be caused by the identified issue in terms of the results of the containment ageing management program, but this may be the result of design changes (for example, implementation of elements to cope with severe accidents), or scheduled maintenance of individual elements of the building.

7.2 Experience of the licence holder with the application of AMP for concrete containment structures

The periodic monitoring of tightness and strength of the containment, as well as the geodetic measurement and assessment of settlement of the main generating unit building are carried out since the units were put into operation. These measurements provide the initial information about the state of providing for the design functions of the containment. A systematic approach to the ageing management program for the containment was developed by the licence holder and gradually introduced into practice approximately in 2008. The overall concept and preparatory analyses were done still before the introduction of the AMP for the containment. An expert external organization in cooperation with the specialists of the licence holder within the preparation of AMP, has developed a range of inspections for individual building structures, including proposal for the interval of its repetitions. In the framework of the first building and technical survey, check points were set for periodic inspections. Since 2008, a specified program of inspections for concrete structures of the containment has been repeatedly implemented on the operated units. Based on the results of the checks, recommendations were proposed that were applied by the licence holder.

With the participation of the representatives of the licence holder on the IGALL and OECD project, the scope of the AMP for the containment and the experience of its implementation were discussed with the representatives of other participating operators of NPPs. On the basis of comparison it can be stated that the overall concept of AMP, as well as the scope of inspections, were set up similarly to other NPPs and no significant suggestions for improvements were identified.

In carrying out inspection of individual building structures of the containment, there is an attempt on the part of the licence holder, to use new progressive methods of inspection. As part of this approach, inspection of steel reinforcement by geo-radar has been successfully implemented. This inspection allows non-destructive tests of the possible corrosion attack of the concrete reinforcement, as well as check of its correct distribution.

At present, the possibilities of refining the non-destructive measurements of depth of cracks in concrete are analysed.

7.3 Assessment and conclusions by the regulator on ageing management of concrete containment structures

The containment AMP has been implemented by the licence holder and is carried out in accordance with the guide [86]. This guide is valid for the operated nuclear units in Slovakia, NPP EBO V2 and NPP EMO. For the NPP units MO3&4 in the construction phase, the AMP for the containment will become effective before they are put into operation. The periodic monitoring of tightness and strength of the containment, as well as geodetic measurement and evaluation of the settlement of the main generating unit building, are carried out since the NPP units were put into operation.

After refuelling of each operating unit and before its start-up, the results of in-service inspections of classified equipment are assessed with the participation of ÚJD SR. Then the licence holder sends reports to the ÚJD SR from the preliminary assessment of results a from evaluation of results of in-service inspections of classified equipment in a form of regular reports according to the ÚJD SR´s Decision [3] and the ÚJD SR issues its opinion on these. In-service inspections of classified equipment are carried out by the licence holder according to the programs of in-service inspections approved by the ÚJD SR. In addition to evaluating results of in-service inspections of classified equipment, the licence holder also notifies to ÚJD SR the extent of work on the containment, for its maintenance and improvement of its tightness, and the results of the tightness test of the containment. This information is also included in the regular reports. ÚJD SR inspectors carry out planned inspection on periodic integral tightness tests of the containment carried out once in two years, where they attend these tests, and they also check the documentation prescribed by the ageing management programs for the containment. In those years, when the periodic integral tightness test of the containment is not carried out, the licence holder performs local measurements of tightness, the results of which are submitted to the ÚJD SR and the ÚJD SR issues its opinion. Once every ten years the licence holder performs the integral strength test of the containment, to which he invites the ÚJD SR and the results of which are submitted to the ÚJD SR according to the Decision [3] and the ÚJD SR issues its opinion. In addition to planned inspections as mentioned above, the ÚJD SR conducts also unplanned inspections in response to various events in the world. Ageing management is also subject of inspections for periodic safety review.

In most cases, inspections were without negative findings, and in case of a negative finding, the ÚJD SR issues a Protocol to the licence holder containing corrective actions and their deadlines, and subsequently checks fulfilment of such corrective actions within the specified deadlines. The inspections did not identify any major deficiencies that would require immediate corrective action. Capability of the SSC relevant for nuclear safety to fulfil their safety functions is ensured.

8 Pre-stressed concrete pressure vessels (AGR)

The chapter is not applicable for Slovakia. Pre-stressed concrete pressure vessels are not being used in the nuclear installations in Slovakia.

9 Overall assessment and general conclusions

Ageing management and lifetime assessment started to be implemented at the NPPs from 1991, being part of several projects aimed at increasing nuclear safety and reliability of operation [64].

The rules of systematic approach to ageing management of SSC are legislatively defined in the ÚJD SR documents [7], [5] and [94]. These documents are based on the recommendations of the IAEA safety requirement [9] and safety guide [1]. Ageing management is one of the areas under the periodic safety review [7].

The basic legislative requirements are reflected in the IMS process documentation of the licence holder [14] and in the relevant proactive ageing management programs (i.e. with foresight and anticipation) developed for SSCs relevant in terms of nuclear safety. The ageing management process is implemented on the operated units of NPP EBO V2, NPP EMO, as well as NPP units MO3&4 under construction.

SE, a. s. has a Lifecycle Management Group – Nuclear, to implement ageing management. In the process model of the licence holder the ageing management is included in the IMS top process as a part of "Generation, Process Engineering".

The cable ageing management program (AMP) is implemented by the licence holder and is performed in accordance with the guide [67]. This guide is valid for all nuclear units in Slovakia, i.e. the operated NPP EBO V2, NPP EMO and NPP units MO3&4 under construction. The individual sub-programs within the cable AMP (surveillance specimen program, functional cable measurements in operation) cover the main degradation mechanisms identified based on operational experience and international recommendations. The licence holder monitors also parameters of the environment (temperature, radiation dose, relative humidity), to which the cables in operation are exposed to. Monitoring includes areas of the containment, and also outside of the containment on both operated power plants.

Ageing management of concealed pipes is a part of AMP for essential service water pipelines [76] that is implemented by the licence holder. This guide is valid for operated NPP EBO V2, NPP EMO. For the NPP units MO3&4 under construction, the AMP will be put into effect before their start. The scope of activities of ESW AMP (corrosion monitoring, concrete monolith monitoring, wall thickness measurement, visual inspections) covers the monitoring of all relevant degradation mechanisms identified on the basis of operational experience, international recommendations and results of ageing management program. Reconstruction and replacement of these pipes was carried out based on monitoring of the status of ESW piping at NPP EBO V2.

The ageing management program for the reactor pressure vessel (RPV) is implemented by the licence holder and has been performed in accordance with the guide [83]. This guide is valid for all nuclear units in Slovakia, i.e. operated NPP EBO V2, NPP EMO and units of NPP MO3&4 under construction. The scope of activities within AMP for RPV (surveillance specimen program, fluency monitoring, assessment of fatigue damage, in-service inspections) covers monitoring of all relevant degradation mechanisms identified based on operational experience, international recommendations and results of ageing management program. Surveillance specimen program was extended with new materials located in the reactor core of RPV and covers the operating conditions with increased power of nuclear units and use of a new type of nuclear fuel for a period of 60 years. The OSART mission in 2010 at NPP EBO V2 identified this program as "Good Practice".

The ageing management program for the containment of the NPP is implemented by the licence holder and has been performed in accordance with the guide [86]. This guide is valid for the operated nuclear units in Slovakia, NPP EBO V2 and NPP EMO. For units of NPP MO3&4 in the phase of construction the AMP for the containment will be put into effect before their start. The periodic monitoring of tightness and strength of the containment, as well as geodetic measurements and assessment of the settlement of the main generating unit have been carried out since the start of operation of the NPP units.

In connection with the long-term operation program, the SSC ageing management programs are developed for a period of 60 years of operation, which the WANO PEER REVIEW mission at the NPP EMO in operation conducted in 2013, and OSART mission (extended with the LTO module) on NPP EBO V2 in operation conducted in 2010, classified as "Good Practice".

The licence holder, according to the Decision [3] submits to ÚJD SR regular reports about the results of ageing management programs, focusing on:
- Lifetime use of the reactor pressure vessel and selected unit equipment, including critical temperature of the brittle fracture of RPV,
- Assessment of critical temperature of RPV brittleness based on chain test for RPV surveillance specimens,
- Evaluation of in-service inspection programs.

The licence holder is involved in the following international projects:
- The licence holder is a member of the IAEA IGALL Project – a project aimed at developing a practical manual for the ageing management of NPP equipment relevant for nuclear safety, including recommendations for effective management of ageing management programs. The Project is divided into three professional fields: machinery equipment, electrical equipment and building structures of NPPs. The licence holder is represented in each working group.
- OECD/NEA CADAK – a project aimed at expanding existing databases and knowledge in the field of ageing management and cable qualification. The licence holder has been participating in this project since 2011.
- OECD/NEA CODAP – a project aimed at extending existing databases with failures of passive components of the primary circuit, as well as other components, whose failure has a significant impact on operation, including measures from the point of degradation mechanisms. The licence holder is provided with access to the Operational Experience Database, as well as the ability to contribute information about occurrence of events in operated nuclear power plants.
- OECD Halden Reactor Project – Fuels and Materials – a project aimed at development of the Small Punch Test methodology, which allows the determination of basic mechanical properties through small samples. The aim of the project was to directly correlate the results with the currently solved surveillance specimen programs for NPP EBO V2 and NPP EMO, and to assess the impact of irradiation of samples in the power and research reactor on the rate of degradation of mechanical properties.
- The licence holder has access to the EPRI (Electric Power Research Institute) databases and materials in the field of ageing management.
- The licence holder is a member of the International Equipment Reliability Working Group, which focuses on the exchange of experience in the Equipment Reliability process.

Slovakia has established a legislative framework for ageing management. The licence holder has the obligation to establish and implement an ageing management program to identify all ageing mechanisms related to systems, structures and components relevant for safety. This (AMP) defines the possible consequences of ageing and sets out the necessary activities to maintain the operability and reliability of these systems, structures and components (SSC).

Regular inspections are carried out by ÚJD SR at the licensee to check the compliance with the legal requirements and international safety standards and best practices. The licensee is encouraged to further implement the AMP by participation in the international projects and to exchange experience with other NPP operators.

During the preparation of the national report the following good practices have been identified:

- In the strategic plans in the field of AM, in the methodology documentation the licence holder addresses not only the issue of obsolescence, but also the development of long-term rehabilitation strategy for SSCs,
- The licence holder maintains a special database for the AM purposes,
- Development of AMP for classified equipment for a long-term operation.

The process also identified challenges:

- Deficiencies in the SSC drawings in relation to the actual status,
- Non-continual updating of AM database to reflect the actual status of SSCs and knowledge.

Capability of systems, structures and components relevant in terms of nuclear safety to fulfil their safety functions is considered by ÚJD SR as secured.

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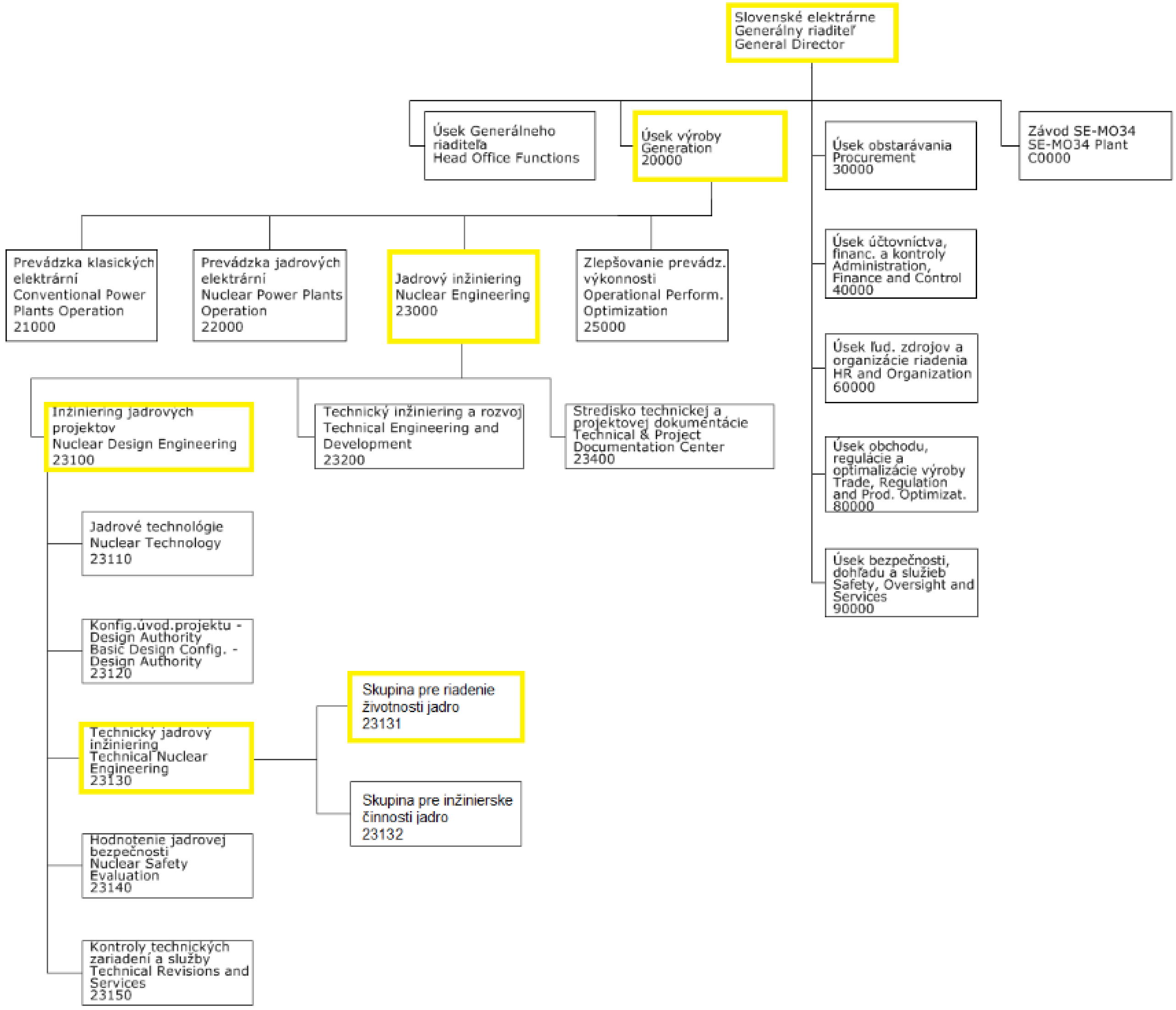
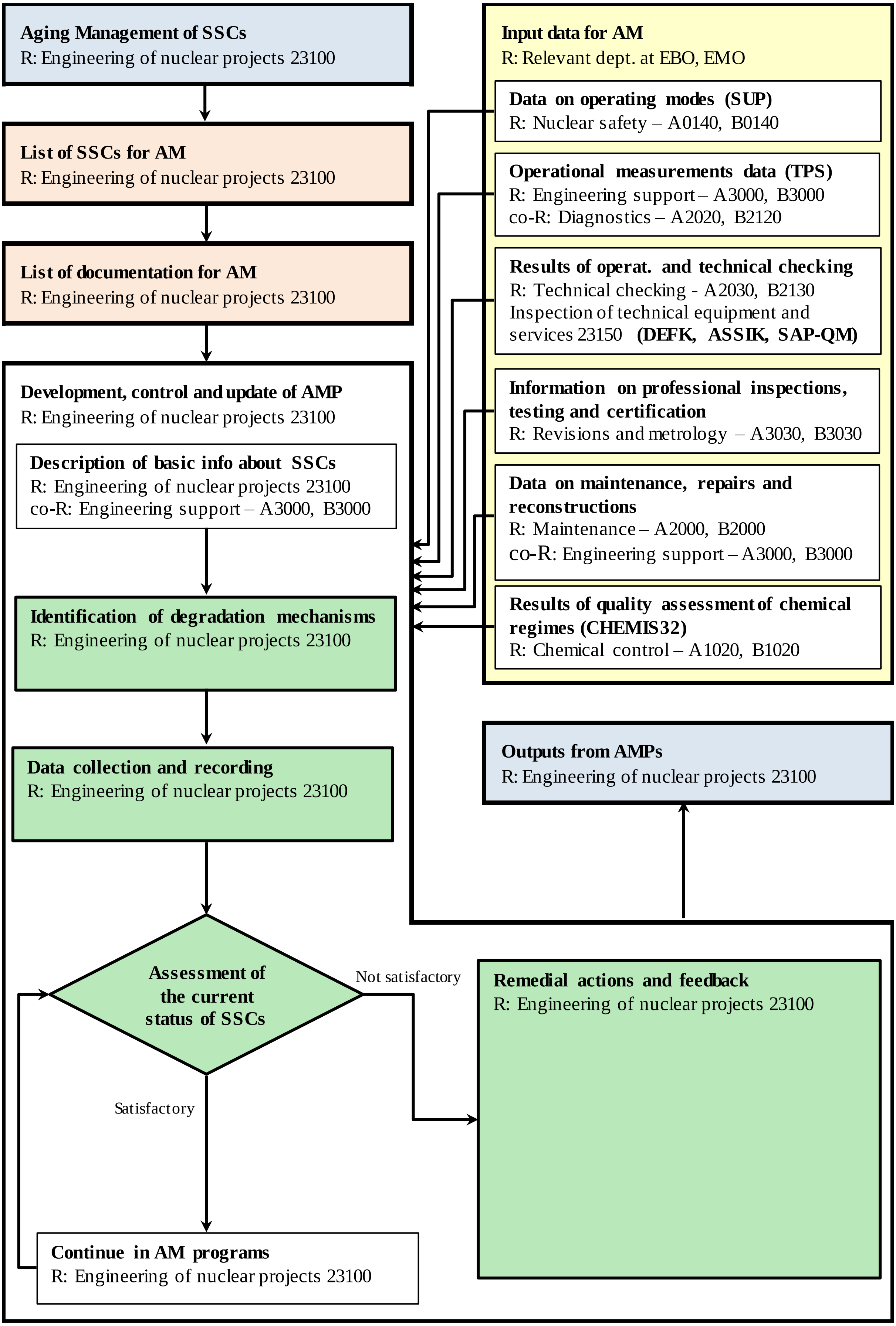
Annex 1 Organizational placement of Lifecycle Management Group



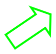
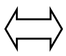
Annex 2 Detailed description of activities in ageing management



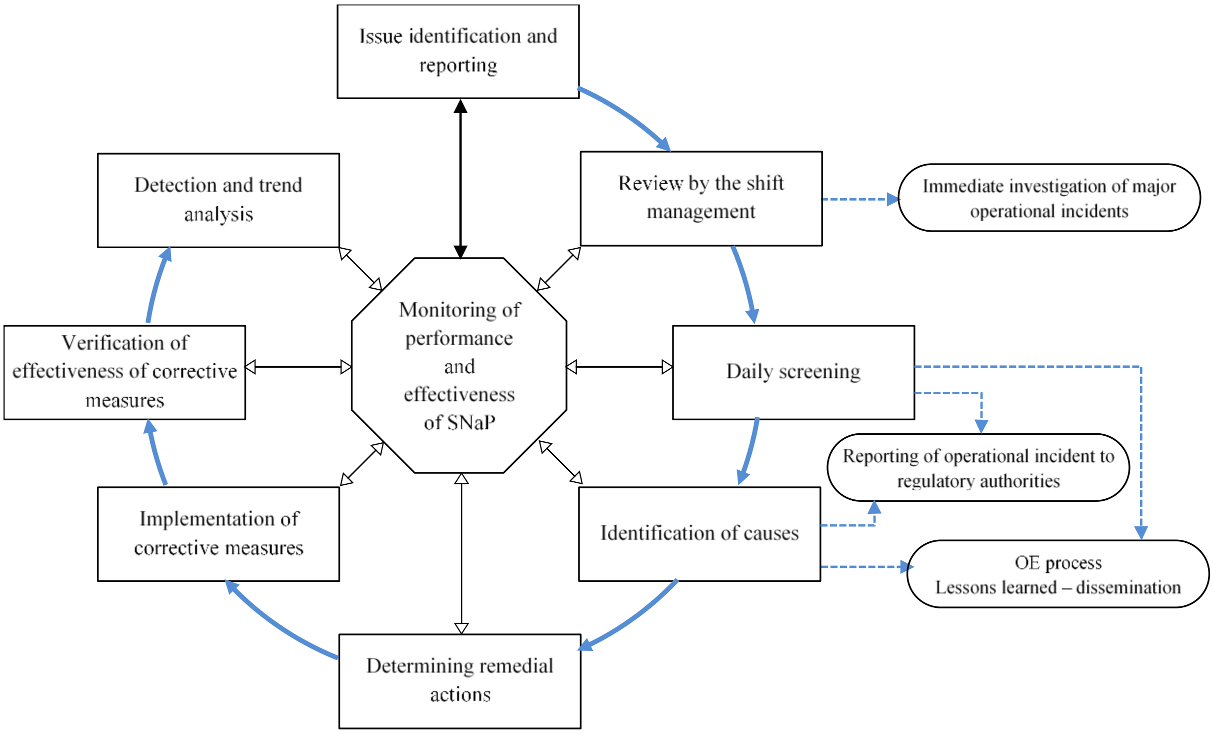
Annex 3 List of AMPs

Identification	Name
JE/NA-311.09-02	Ageing Management Program – Reactor pressure vessel
JE/NA-311.09-03	Ageing Management Program – Steam generators
JE/NA-311.09-04	Ageing Management Program - Main circulating water pumps
JE/NA-311.09-05	Ageing Management Program – Main gate valves
JE/NA-311.09-06	Ageing Management Program – Primary circuit piping
JE/NA-311.09-07	Ageing Management Program - Pressurizer
JE/NA-311.09-08	Ageing Management Program – Secondary circuit piping
JE/NA-311.09-09	Ageing Management Program – Essential service water piping
JE/NA-311.09-10	Ageing Management Program – Main condensers
JE/NA-311.09-11	Ageing Management Program - Cables
JE/NA-311.09-12	Ageing Management Program – Main generating unit including containment
JE/NA-311.09-13	Ageing Management Program – Reactor internals
JE/NA-311.09-14	Ageing Management Program – Monitoring corrosion condition of SSCs at NPP
JE/NA-311.09-15	Ageing Management Program – Diesel generator station
JE/NA-311.09-16	Ageing Management Program - Central pumping station
JE/NA-311.09-17	Ageing Management Program – Fan cooling towers
JE/NA-311.09-18	Ageing Management Program – Chemical water treatment NPP EBO V2
JE/NA-311.09-19	Ageing Management Program – Auxiliary building
JE/NA-311.09-20	Ageing Management Program – Turbine hall and foundations of TG

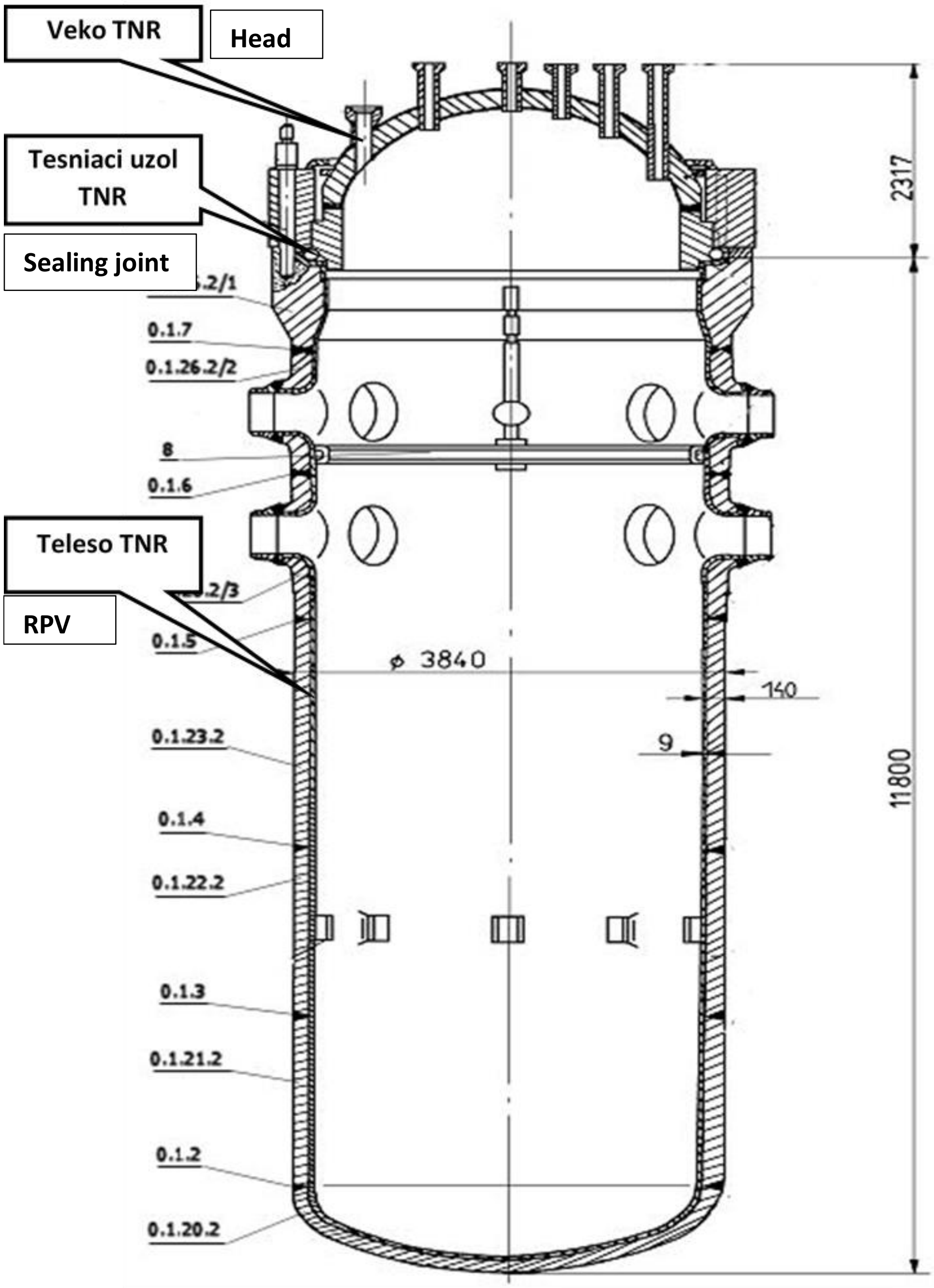
Annex 4 Colour evaluation system and performance trending

Colour evaluation system and performance trending with brief definitions		
EVALUATION	**PERFORMANCE**	**ACTION**
GREEN	Good	It does not require increased attention
WHITE	Acceptable	Current performance/activities are reasonable
YELLOW	Conditionally acceptable	It requires increased attention
RED	Not acceptable	It requires increased monitoring and remediation
TREND	**PERFORMANCE**	**DESCRIPTION**
	Improving	It means an improvement with the colour change over the past period
	Trend of improvement	It means an improvement without changing the colour over the past period
	Steady	No change in the condition for the past period
	Worsening trend	It means worsening of condition without changing the colour over the past period
	Worsening	It means worsening of condition with colour change over the past period

Annex 5 Basic scheme of SNaP process



Annex 6 Reactor pressure vessel scheme



Annex 7 Containment with vacuum-bubble condenser