Státní úřad pro jadernou bezpečnost

NATIONAL ASSESSMENT REPORT OF THE CZECH REPUBLIC

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

Prague 2017

National Report of the Czech Republic for the Purposes of Topical Peer-Review "Ageing Management" under the Nuclear Safety Directive 2014/87/EURATOM

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Preamble

This report has been prepared for the purposes of the first Topical Peer-Review (hereinafter referred to as the "TPR"), which arises from the European Union's Nuclear Safety Directive 2014/87/EURATOM. The Directive requires undertaking the TPR thereunder every six years with the first starting in 2017. It has been decided that the topic for the first TPR is "Ageing Management". The objective of this Peer-Review is to undertake a peer review of practices and approaches in the area of ageing management, to identify strengths and weaknesses or good practices and to define areas for improvement, to share operating experience and also to provide a transparent and open framework for developing and implementing appropriate follow-up measures to address areas for improvement. The TPR includes all nuclear power plants and research reactors with a thermal power equal to 1 $MW_{t,}$ or more that will be operating on 31 December 2017 or under construction on 31 December 2016. Research reactors with a power below than that stated above may also be included on a voluntary basis.

The groups of components were then set as examples of the implementation of the overall Ageing Management Programme, of which the following groups fall within the scope of the TPR for the Czech Republic: electrical cables, concealed pipework, reactor pressure vessels and concrete containment structures.

The first task of the peer review was to draw up this National Assessment Report. This shall be followed by a peer review of the national reports of each Member State in the form of questions and answers on the information referred to in each report. The whole process shall be completed by conducting a peer-review workshop, publishing a report on that workshop and setting an implementation plan for remedial measures arising from the entire assessment.

The report has been prepared in accordance with the Technical Specifications for National Assessment Reports [1] formulated by the RHWG WENRA and subsequently confirmed by the WENRA as well as the ENSREG. The Technical Specifications determine the desired outline and content of national assessment reports.

The principle objective of the national assessment report is to collect information concerning the selected topic on the basis of which the peer review can be carried out. Specifically, this is a description of the so-called "overall" ageing management programme focusing on programme aspects of the ageing management process, implementation of that overall ageing management programme and experience with the application of ageing management. The descriptive part is followed by the evaluation of compliance with the national and international requirements, identification of the strengths and weaknesses of the process and definition of the areas for improvement. The purpose is to provide sufficient detail to enable all participating countries to carry out a meaningful peer review.

The report does not contain any sensitive information subject to export control of dual-use items.

The assessment, on the basis of which the report was produced, was carried out on 30 June 2017. In the event of changes in any of the facts referred to in the report from that date to the report publication date, such differences shall be provided in the national presentation in the peer-review workshop.

List of abbreviations

AM	Ageing management
AMP	Ageing Management Programme
AMPC	Component Specific Ageing Management Programme for Cables
AMR	Ageing Management Review
BPIG	Buried Pipe Integrity Group
BPIRD	Buried Pipe Inspection Results Database
CCW	Circulation Cooling Water
CDFM	Conservative Deterministic Failure Margin
CEO	Chief Executive Officer
CRDM	Control Rod Drive Mechanism
CTU	Czech Technical University Prague
CV	Centrum výzkumu Řež s.r.o. (Research Centre Řež) - operator of the LVR-15 research
ČEZ	ČEZ, a. s. - operator of the Dukovany NPP and the Temelín NPP
ČSN	Czech National Standard
DBA	Design basis accident
DC	Direct current
DCVG	Direct Current Voltage Gradient
EBO	Bohunice NPP (Slovak Republic)
EC	European Commission
ECCS	Emergency Core Cooling System
EDMET	Electrodiagnostics of magnetic pipes
EDU	Dukovany NPP
EMC	Electromagnetic compatibility
EMO	Mochovce NPP (SK)
EN	European Standard
ENR	Event Notification Report
ENSREG	The European Nuclear Safety Regulators Group
EPRI	Electric Power Research Institute
EPS	Emergency power supply
ESW	Essential service water
ETE	Temelín NPP
EU	European Union
EURATOM	European Atomic Energy Community
FSAR	Final Safety Analysis Report
HELB	High-energy Line Break
HR	Health Report
HV	High Voltage
HZ	Hermetic zone
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IASCC	Irradiation-assisted stress corrosion cracking
IGALL	International generic ageing lessons learned
IGSCC	Intergranular stress corrosion cracking
IQAP	Individual quality assurance programmes

IRS	International Reporting System for Operating Experience
ISI	In-service inspection
JIT	Just-in-time
LaC	Limits and Conditions of Safe Operation
LOCA	Loss-of-coolant accident
LTO	Long Term Operation
LV	Low voltage
MCP	Main circulation pump (main coolant pump)
MIV	Main isolation valve
MKS	Modular control system
NDT	Non Destructive Testing
NEA	OECD Nuclear Energy Agency
NFM	Neutron flux measurement
NESW	Non-essential service water
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission (USA)
NS	Nuclear safety
NTD	Normative Technical Documentation
NUGENIA	Nuclear Generation II & III Association – non profit organisation
OECD	Organisation for Economic Co-operation and Development
OP	Operational Programme
PCS	Protection and Control System
PERIZ	Periodic Integral Tightness Testing
PIE	Postulated initiating event
PJB	Pre-job-briefing
PM	Preventive maintenance
PMBD	Preventive maintenance basis database
PSA	Probabilistic safety assessment
PSR	Periodic safety review
PT	Penetration testing
PTS	Pressurized Thermal Shock
R&D	Research and Development
RHWG	Reactor Harmonisation Working Group
RPV	Reactor Pressure Vessel
SALTO	Safe Long Term Operation
SBO	Station Blackout
SC	Safety Class
SCC	Stress corrosion cracking
SCs	Structures and components
SE	Slovenské elektrárne (operator of the Mochovce NPP and the Jaslovské Bohunice
SG	Steam generator
SRL	Safety Reference Level
SRS	Safety report Series
SSCs	Systems, structures and components
SSG	Specific Safety Guide
SSK	Cabling management system

SSSP	Supplementary Surveillance Specimen Programme
SÚJB	State Office for Nuclear Safety
SW	Software
TGSCC	Transgranular stress corrosion cracking
TLAA	Time Limited Ageing Analysis
TM	Temperature Measurement
TOFD	Time of Flight Diffraction (ultrasonic testing method)
TPR	Topical Peer Review
TS	Technological system
UT	Ultrasonic testing
UTT	Ultrasonic thickness testing
VT	Visual testing
VVER	Water-cooled and water-moderated reactor (Russian design)
VVK	Life calculation and evaluation system of EDU and ETE safety cables
WANO	World Association of Nuclear Operators (*WANO*
WENRA	Western European Nuclear Regulators Association
WER	WANO event report

1. General information

1.1 Nuclear installations identification

The following nuclear installations with nuclear reactors are operating in the Czech Republic on 31 December 2017:

Site	Holder of license	Nuclear installation	Number of reactors	Type	Reactor power output	Year of first operation	Included in the assessment
Dukovany	ČEZ	Dukovany NPP	4	VVER 440/213	4 x 500 MW_e 4 x 1444 MW_t	1985-1987	Yes
Temelín	ČEZ	Temelín NPP	2	VVER 1000/320	2x 1080.3 MW_e (at 18.5°C CCW) 2x 3120 MW_t	2004	Yes
Řež	CV Řež	Research reactor in Řež	1	LVR-15	Below 10 MW_t	1957 (1989)	Yes
	CV Řež	Research reactor in Řež	1	LR-0	0 MW_t	1982	No
CTU Prague	CTU	Research reactor in Prague	1	VR-1	0 MW_t	1992	No

The following installations are included in the assessment for the purposes of the TPR in the Czech Republic: Dukovany Nuclear Power Plant Units 1 – 4, Temelín Nuclear Power Plant Units 1 and 2, and LVR-15 research reactor, with its thermal output exceeding 1 MW_t. Other nuclear installations do not meet the defined scope of installations to be assessed herein. Research reactors LR-0 and VR-1 nuclear research reactors have not been included in the assessment on a voluntary basis because of significantly lower output than 1 MW_t.

The fixed date for their shutdown is not set for any of the above listed nuclear installations.

As from 1 January 2018, all units of the Dukovany Nuclear Power Plant are assumed to be in a state of the so-called “long-term operation” (LTO). The residual life assessment of systems and components important to safety and the continuously implemented Equipment Refurbishment Programme show that the conditions for safe operation for at least next ten years, with a view to the operation until 2035 (2037 respectively), are met.

Current status of SÚJB Decisions on Operation (license to continue operation) of the Dukovany NPP (issued for indefinite time period):

Reactor unit 1 - SÚJB Decision on Operation was issued on 30 March 2016

Reactor unit 2 - SÚJB Decision on Operation was issued on 29 June 2017

Reactor unit 3 - SÚJB Decision on Operation was issued on 19 December 2017

Reactor unit 4 - SÚJB Decision on Operation was issued on 19 December 2017

The Temelín Nuclear Power Plant currently has its license for operation valid until 12 October 2020 (Unit 1) or 31 May 2022 (Unit 2).

The nuclear research facilities have their existing licenses valid until:
LVR – 15 until 31 December 2020
LVR – 0 until 31 December 2020
VR – 1 indefinite

1.1.1 Description of the nuclear installations included in the assessment

Dukovany Nuclear Power Plant

The Dukovany Nuclear Power Plant consists of four reactor units VVER-440/213 in the form of two twin units; each unit has its own reactor building with the common reactor hall. The individual units are of the identical technical design. The installed power output is 4 x 510 MW_e, the thermal output of the individual reactors of the Dukovany NPP is 1,444 MW_t. The VVER-440/213 reactors belong to pressurized water reactors of Generation II. The reactors were put into operation in 1985 (Unit 1), 1986 (Unit 2) and 1987 (Units 3 and 4). Nuclear safety assurance for the VVER-440/213 design is based on the multiple barriers preventing the release of radioactive material, including containment and the concept of multiple redundancies of safety systems.

The reactor (or the reactor core) is cooled and moderated by water of the primary circuit. The water is circulated through the core by means of main circulation pumps. After flowing through the reactor, the heat accumulated in the coolant is transferred to secondary circuit water in the steam generators. The reactor core consists of 312 fuel assemblies and 37 control rod assemblies arranged in a hexagonal pattern. The solution of chemically treated water and boric acid serves as a reactor coolant and moderator, and the low enriched uranium oxide ^{235}U serves as fuel. The fuel elements are arranged in a hexagonal lattice. The chemical control of chain reaction is ensured by means of boric acid and the mechanical regulation is based on the “tandem” system (the fuel assembly is gradually withdrawn from the core while inserting the absorber cylinder).

The reactor coolant system (primary circuit) consists of six loops (pipes of DN 500). Each of the loops is fitted with the main circulation pump (MCP), horizontal steam generator (SG) and two main isolation valves (MIV). The MIVs allow isolate leaking main circulation piping or SG of the loop in question. The primary circuit also includes the pressurizer system, which maintains the pressure of the primary circuit. The reactor pressure vessel and the primary circuit are designed for the pressure of 13.729 MPa at the temperature of 350°C, with the rated pressure and temperature at reactor outlet of 12.261 MPa and 297.2°C, respectively.

The reactor and the main components of the primary circuit are located in containment, which consists of the reinforced concrete structure with the hermetic liner and which is the barrier preventing the release of radioactive material to the environment. The containment is located inside the reactor building, which continue above the main level by the steel structure forming its roofing. The containment is designed for the design pressure of 150 kPa and for the temperature of 127°C.

In the reactor building, there are spent fuel storage pools where spent fuel from the core is being stored. When residual heat decreases to acceptable level, the fuel in the CASTOR containers is transported to the interim spent fuel storage or the spent fuel storage on a continuous basis.

The secondary circuit consists of two turbo-generators for one unit, as well as the condensation, regeneration, feedwater and steam line systems. The steam generators produce steam with the pressure of 4.751 MPa and the temperature of 260.7°C, which drives the pair of steam turbines. The turbine drives the generator; after transformation into the extra high voltage, the electricity is supplied into the grid. After transfer of its energy, the steam comes from the turbine into the condenser, where, after cooling, it condensates and returns back through the regeneration system into the steam generators. The turbine building is always common for two reactor units. Therefore, there are four turbo-generators alltogether at this building.

The secondary circuit is followed by the circulation cooling water systems with four cooling towers per one main generating unit, i.e. twin reactor unit and three independent service water systems, which are cooled by the newly built ultimate heat sink (forced-draught cooling towers).

Residual heat is removed to the atmosphere during the operation by circulation water through the main condensers and cooling towers.

During the outage, the unit is cooled through the steam generators, technological condensers, and essential service water system (ESW). Heat removal from the essential service water system to the atmosphere is realized through the forced-draught cooling towers of the ultimate heat sink, while the possibility still remains to use the ESW cooling by means of cooling towers. The essential service water pumping station, together with the circulation water pumps, non-essential service water (NESW) pumps and fire water pumps, is designed as an independent civil structure for the one main generating unit (i.e. twin unit); which means that there are two ESW pumping stations in the nuclear power plant's premises in total.

The active safety systems (high-pressure, low-pressure and spray) have the 3 x 100 % redundancy and are mutually independent and physically separated. The passive safety systems (passive accumulator tanks with boric acid - the hydro accumulators inside the containment) have the 2 x 100 % redundancy. Seismic resistance of all the redundant safety systems, including the power supply and the control systems as well as all the other auxiliary systems is assured. Emergency power supply systems and the safety related instrumentationcontrol systems are independent of each other, physically separated and seismic-resistant (they are subject to qualification applicable to safety systems).

The design disposes of diversified systems for the fulfilment of three fundamental safety functions:

1) control of reactivity (reactor shutdown)
2) heat removal from reactor and fuel store
3) confinement of radioactive material (barriers and the containment's isolation).

Emergency core cooling system components together with spray system components are located at the bottom levels of the reactor building. Three independent spray systems are connected to the SG box in order to ensure pressure reduction in the hermetic zone if necessary.

The on-site power system of each unit includes the transformers for own consumption, 6 kV and 0.4 kV switchgears, DC power equipment with the 220 V and 48 V batteries and the protection system, process control system, and the alarm system of the main equipment and on-site power supply. The output of the Dukovany NPP is led out by means of the 400 kV line to the Slavětice switch yard approximately 3 km far away from the Dukovany NPP. In addition to the operational on-site power system from the 400 kV line, each unit of the Dukovany NPP has the stand-by on-site power system provided by two 110 kV lines common for two units. In addition, each unit is equipped with three diesel generators, which serve as independent emergency power supply units (3x100%) in case of the loss of operational and stand-by power supply. Two SBO diesel generators common for all units and independent of standard equipment, fuel, cooling and power supply were added to the standard equipment of three independent diesel generators for each generating unit. All the stand-by power supply systems and the control systems are independent of each other, physically separated and seismic-resistant.

General layout of the Dukovany NPP is shown in Fig. A.1 in Annex A.

Temelín Nuclear Power Plant

The Temelín Nuclear Power Plant comprises two nuclear units with VVER-1000 pressurized water reactors of the V 320 series type, each with the installed power output of 1080.3 MW_e and the thermal output of one unit amounts to 3,120 MW_t. The VVER-1000/320 reactors belong to pressurized water reactors of Generation II. The reactors were put into operation in 2004. As in the case of the Dukovany NPP nuclear safety assurance of the VVER-1000/320 design is based on the multiple barriers preventing the release of radioactive material, including containment and the concept of multiple redundant safety systems.

The reactor is cooled and moderated by primary circuit water, pumped through the core by means of main coolant pumps. After passage through the reactor, the heat accumulated in the coolant is transferred to secondary circuit water in the steam generators. The reactor core consists of 163 fuel assemblies and 61 control rod assemblies arranged in a hexagonal pattern. The solution of chemically treated water and boric acid serves as a reactor coolant and moderator, and the low enriched uranium isotope ^{235}U serves as fuel.

The pressure in the primary circuit is maintained by the pressurizer. The reactor coolant system (the primary circuit) consists of four loops of circulation piping (DN 850), each fitted with the main circulation pump and the horizontal steam generator. The reactor pressure vessel and the primary circuit are designed for the pressure of 17.6 MPa at the temperature of 350°C (the operating pressure is 15.7 MPa at the temperatures of 290 - 320°C).

The primary circuit equipment is located inside the containment made from pre-stressed concrete. The containment consists of the cylindrical structure made from pre-stressed concrete with the inner diameter of 45 m, closed by the hemispherical cover. The inner surface of the containment is covered by hermetically tight steel liner. The containment is designed for the design pressure of 0.49 MPa and for the design temperature of 150°C.

The spent fuel removed from the reactor core is placed into the spent fuel storage pools, which are located inside the containment. When residual heat decreases to acceptable level, the spent fuel is put to the cask and taken to the spent nuclear fuel storage (its capacity corresponds with the plant lifetime).

The secondary circuit consists of the steam generators, feedwater system, one turbo-generator with the rated electrical output of 1080.3 MWe, condensation system, and regeneration system. The steam generators generate the steam with the pressure of 6.32 MPa and temperature of 279°C, which drives the steam turbine with the capacity of 1080.3 MWe.

Residual heat is removed to the atmosphere during the operation by the circulation cooling water system through the main condensers and cooling towers.

After the reactor shutdown, residual heat of the core is removed to the atmosphere from the turbine condenser by means of the cooling towers. When the core temperature drops below 150°C, this residual heat is removed by means of heat exchangers of the low-pressure emergency system to the essential service water cooling circuit, which also cool down the equipment important to nuclear safety; the essential service water is cooled in special outdoor pools (cooling pools with the spray system).

The safety systems, composed of three independent divisions separated from each other from construction and electrical point of view, serve the purposes of emergency reactor cooling and pressure reduction in the containment. One division in operation is sufficient for emergency management. Each division contains tanks of the emergency core cooling system, emergency boric acid storage tanks, emergency cooling spray pumps, high-pressure and low-pressure emergency cooling pumps and other components.

The active safety systems have the 3 x 100 % redundancy and are mutually independent and physically separated. The passive safety systems (the hydro accumulators inside the containment) have the 2 x 100 % redundancy. Seismic resistance of all the redundant safety systems, including the power supply and the control systems as well as all the other auxiliary systems is assured.

The on-site power system of each unit includes the auxiliary normal transformers, 6 and 0.4 kV switchgears, DC power equipment with the 220 V batteries and the protection system, process control system, and the alarm system of the main equipment and on-site power mechanisms. The output of the Temelín NPP is led out by means of the 400 kV line to the Kočín switch yard. In addition to the operational on-site power system from the 400 kV line, each unit of the Temelín NPP has the stand-by on-site power system provided by 110 kV line. In addition, each unit is equipped with three diesel generators, which serve as independent emergency power supply units (3x100%) in case of the loss of operational and stand-by power supply. Additional two diesel generators are common for both units, which serve as independent emergency power supply for the equipment related to nuclear safety. Two SBO diesel generators common for both units and independent of the existing equipment, fuel, cooling and power supply were added to the standard equipment of the diesel generators. All the stand-by power supply systems and the control systems are independent of each other, physically separated and seismic-resistant.

General layout of the Temelín NPP is shown in Fig. A.2 in Annex A.

LVR-15 research reactor

The LVR-15 research reactor is situated in the premises of ÚJV Řež, a.s. in Řež u Prahy. The license holder is the company Centrum výzkumu Řež s.r.o. (Research Centre Řež). The reactor in its current form has been in operation since 1989, when the original VVR-S reactor, operated from 1957, was modernised. Among other things, the main components of the primary circuit including reactor vessel were replaced in the modernisation.

The LVR-15 research reactor is the research tank-type light water reactor placed in the stainless-steel pressureless vessel under the shielding cover, with the forced cooling, with the IRT-4M type fuel and with the operating thermal output up to 10 MW. The reactor is operated in campaigns. The reactor is normally in continuous operation for 3 weeks followed by a 10–14-day break for maintenance and refuelling or in non-standard campaigns for short-term experiments in accordance with the requirements of the experimenters. The demineralized water is both moderator and coolant, and the reflector is composed of water or beryllium blocks depending on the operational configuration.

The core of the LVR-15 research reactor consists of the aluminium basket (the so-called "separator"), in which the fuel assemblies, beryllium blocks, aluminium displacers and irradiation channels are placed. The centre of the core is located approximately 1.4 m above the reactor bottom. The reactor core lattice is arranged in a rectangular shape of 8 × 10 cells;28–32 cells usually contain fuel assemblies. The control rods are in 12 fuel assemblies. Some cells between the fuel are designed for probe channels. The active channels of experimental loops, the rotary channels for silicon irradiation, the pneumatic tube mail, and the vertical irradiation channels are normally located at the periphery of the core. Other cells contain beryllium reflectors or water displacers.

In the LVR-15 research reactor, the IRT-4M type fuel of 19.7% ^{235}U enrichment from Russia is used. Before 1998 the IRT-2M type fuel of 80% enrichment was used, followed by fuel of 36% enrichment until 2011. The transition to the IRT-4M fuel of 19.7% ^{235}U enrichment was completed in October 2011. The fuel assemblies are of a sandwich type, the core is made of the dispersion of UO_2 and aluminium powder. The fuel assemblies have the form of square tubes, which are concentrically arranged in six- or eight-tube assemblies. The fuel assembly is fitted with aluminium end fittings on both sides. The fuel cladding is also made from aluminium (SAV-1 alloy). The fuel assembly is 880 mm long and the active (fuel) part is 600 mm long. The channel with the control rod can be installed in the central tube.

In order to control the fission chain reaction on the LVR-15 research reactor, 12 control rods are used, which are suspended from the console attached to the support for the vessel in the upper part of the vessel. The boron in the control rods is the neutron absorber on the LVR-15. Of the total number of 12 control rods, there are eight shim rods, three safety rods and one rod is in the automatic regulator mode. The absorbing part of control rods is made from boron carbide (B_4C).

The heat generated in the core of the LVR-15 research reactor is removed to the Vltava River through three cooling circuits. The primary coolant circuit is fitted with five main circulation pumps and two emergency residual heat removal pumps connected to the accumulators, which provide the flow of cooling demineralized water through the core and heat exchangers. In case of loss of off-site power, the reactor core is cooled-down by means of one main pump and one emergency residual heat removal pump. Each of them is powered from a separate diesel generator. The maximum coolant temperature at the reactor outlet is 56°C; the inlet temperature is 45°C, and the maximum coolant flow through the primary circuit is 2,000 m^3/h.

The LVR-15 research reactor is a general purpose nuclear research facility for the needs of the Czech research and industry, which is used in the following areas:

- Loop- and probe-based experiments, focusing on the research in materials and physical metallurgy;
- Experiments on the horizontal channels (neutron physics and solid-phase physics);
- Irradiation services (production of radioisotopes, radiopharmaceuticals, and silicon irradiation);
- Neutron activation analysis, neutron radiography;
- Experiments associated with the irradiation of patients using the method of neutron capture therapy.

For the areas above, the reactor is equipped with the following essential experimental facilities:

- High-pressure water loops (RVS-3, RVS-4, BWR-1, BWR-2);
- Irradiation probes (Chouca CH-MT type and 2CT flat probes), vertical irradiation channels wide (with the inner diameter of 62 mm) and narrow (with the diameter of 44 mm);
- Vertical channels with the rotation for neutron doping with the diameter of 115 mm;
- Vertical irradiation channels for the bombardment of IRE targets;
- Pneumatic tube mail for the short-term bombardment of samples;
- Horizontal radial channels 9x;
- Thermal column;
- Hot cells.

General layout of the LVR-15 research reactor is shown in Fig. A.3 in Annex A.

1.2 Process to develop the national assessment report

Preparation of the National Assessment Report began shortly after publication of the preliminary draft Technical specifications [1], containing the requirements for the content of the National Assessment Report, in the first half of 2016. Parties who have been involved in preparing the report: State Office for Nuclear Safety (SÚJB), holder of the license to operate nuclear power plants - ČEZ, a. s., in cooperation with its support organisation ÚJV Řež, a.s., and Centrum výzkumu Řež s.r.o. (Research Centre Řež) as the license holder for the LVR-15 research reactor.

At the request of the SÚJB, guarantors responsible for providing technical information to be included in individual chapters were appointed on the part of ČEZ, a.s. (ÚJV Řež, a.s.) and the Centrum výzkumu Řež s.r.o. (Research Centre Řež), who reported at regular intervals to the SÚJB on the progress of their preparation. The preparation of the report in the SÚJB was coordinated under Instruction No. 5/2016 of the SÚJB Director of Section for Nuclear Safety, which appointed a coordinator for the preparation of the National Assessment Report and a team of authors of individual chapters. In addition, the dates and responsibilities for the process to prepare the report were set therein. Coordination meetings were convened, as appropriate, with holders of license at the level of project coordinators of all stakeholders as well as meetings at the working level, for the purposes of discussion on individual technical topics.

The SÚJB reviewed parts of the chapters developed by holders of license (licensees self-assessment) and added its own assessment of the facts referred to in the documents submitted, which involved the assessment in relation to the requirements, a description of regulatory experience in inspection and assessment activities, and the conclusions. Finally, draft of the National Assessment Report was reviewed by SÚJB´s independent Advisor on Nuclear Safety. Following the settlement of comments arising from this review and also from SÚJB internal review procedure, the report was approved by SÚJB´s Director of Nuclear Safety Division. The Report will be published in Czech and English language versions on the SÚJB public website in January 2018.

2. Overall ageing management programme requirements and implementation

2.1 National legislative and regulatory framework

Requirements on ageing management were included in the legislation documents since nuclear energy utilisation in Czech Republic beginning i.e. in Act No. 28/1984 Coll., Act No. 18/1997 Coll. and their implementing decrees. All this documents were updated and upgraded according to research and development results, operational experience and the growing needs for nuclear safety improvement.

Actually Act No. 263/2016 Coll., “Atomic Act” [2] is the basic legislative document governing the area of peaceful utilisation of nuclear energy and ionising radiation, which entered into force on 1 January 2017. Requirements of EURATOM and EU legislation, e.g. 2014/87/Euratom, have been transposed into that Act. In addition, fundamental internationally recognised principles for the utilisation of nuclear energy and ionising radiation – the so-called “Safety Fundamentals” – i.e. safety standards of the International Atomic Energy Agency, have been incorporated in the Atomic Act.

Implementing regulations - decrees are related to the Atomic Act, elaborating in more detail the requirements of the Atomic Act. In addition, fundamental safety principles and requirements of the IAEA documents of the type “General Safety Requirements” and the requirements of the so-called WENRA “Safety Reference Levels (Criteria)” [3] have been incorporated in the Atomic Act and implementing decrees.

Regulatory safety guides related to the area of nuclear safety and radiation protection are not egally binding, but their observance helps stakeholders to implement the legislative requirements into the practice. They are prepared and published in order to help the addressees of the legal provisions to comply with the legislation and to reach a “good practice”.

The area of ageing management is governed at all three levels of legislative pyramid.

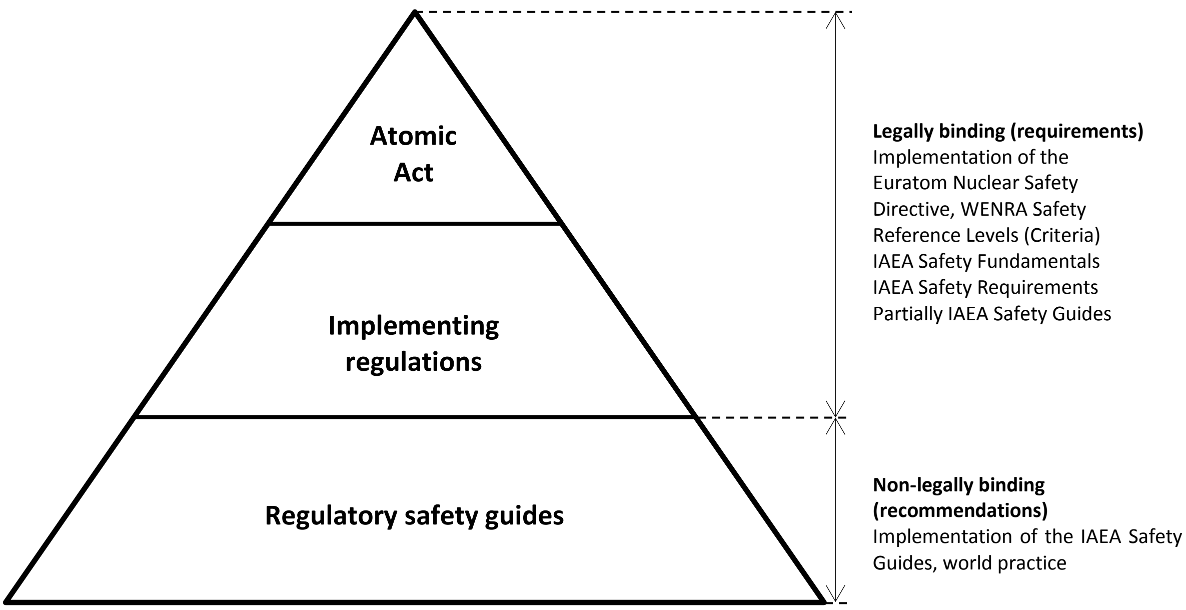

Fig. 2.1: Legislative pyramid

The requirement to establish and maintain the ageing management process implemented according to the Ageing Management Programme at all stages of the lifetime cycle of a nuclear installation, is stated in the Atomic Act.

Detailed requirements for the the content and form of documents related to the Ageing Management Programme are set out in SÚJB Decree No. 21/2017 Coll. [4], on nuclear safety assurance of selected equipment (equipment important to safety; safety classified SSCs) and are based on the internationally established principles of effective Ageing Management Programme. Rules and criteria for the selection of systems, structures and components (SSCs) subject to ageing management process shall be defined in the ageing management process. This selection shall at least include all selected (safety classified) SSCs and safety related SSCs as well. In addition, degradation mechanisms shall be identified for the SSCs included in selection for ageing management (AM) process and the effects of their ageing shall be determined. The holder of license shall take actions to monitor and minimise such degradation mechanisms and ageing effects, shall specify the methods for monitoring and testing for the purposes of timely detection of such phenomena, shall identify the parameters to be monitored and the health indicators including definition of acceptance criteria. The development of ageing effects and the occurence of degradation mechanisms shall be monitored; the SSCs shall be periodically assessed; measures in operation and maintenance shall be taken to mitigate or eliminate degradation mechanisms or ageing effects, and remedial actions shall be taken if the acceptance criteria related to the parameters to be monitored in order to ensure operability and reliability of the SSCs are not fulfiled. All those obligations shall be documented in the Overall Ageing Management Programme, according to which the entire process shall take place, and shall also be reflected in the Ageing Management Programmes at the level of selected components (the so-called “Component Specific Ageing Management Programme”) and possibly in the programmes focused on a specific degradation mechanism or ageing effect (the so-called “Specific Ageing Management Programme”).

For commercial nuclear installations with a power reactor, the legislative pyramid includes the recommendations referred to in Regulatory Safety Guide BN-JB-2.1 “Ageing Management for Nuclear Power Plants” [5]. That Regulatory Safety Guide is based on the recommendation of IAEA Safety Guide NS-G-2.12 “Ageing management” [6] and IAEA SRS No. 57 “Safe Long Term Operation of Nuclear Power Plants” [7], and elaborates the requirements of the relevant WENRA Safety Reference Levels [3]. SÚJB intends to revise Regulatory Safety Guide BN-JB-2.1 soon with the goal to implement requirements of new legislation and guidance given in new revision of the IAEA guide related to ageing management - SSG-48 “Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants”.

For research reactors, relevant recommendations are referred to in Regulatory Safety Guide BN-JB-1.15 “Nuclear Safety, Radiation Protection, Physical Protection and Emergency Preparedness of Research Facilities” [8], which specifies the application of the requirements set in national legislation for research nuclear facilities on the basis of a graded approach.

In addition to explicit obligations directly relating to the ageing management process, the Atomic Act [2] includes the requirement for periodic safety review (PSR), aimed at assessing the extent to which the systems, structures and components, individually or as a whole, of a nuclear installation, including their personnel, meet the current safety requirements set out in Czech legislation, WENRA and IAEA recommendations, and the international practice, and the extent to which the original Design Basis shall continue to apply, on the basis of which the SÚJB issued its Decisions on siting, construction and operation of nuclear installations. A set of measures to maintain and improve safety is the outcome of the periodic safety review, in order to ensure the appropriate level of safety of a nuclear installation throughout the operation until the next periodic safety review, possibly until the end of its lifetime. The assessment of the status of ageing management for SSCs is one of the areas to be assessed. The requirements for periodic safety review are detailed in the SÚJB Decree No. 162/2017 Coll., on requirements for safety assessment under the Atomic Act [9], which defined the areas for assessment, and also Regulatory Safety Guide BN-JB-1.2 [10], which is based on IAEA Safety Guide NS-G-2.10 "Periodic Safety Review of Nuclear Power Plants" [11]. In the context of new legislation, and IAEA SSG-25, that Regulatory Safety Guide shall also be revised.

Other implementing regulations closely related to ageing management of nuclear installations include SÚJB Decree No. 358/2016 Coll., on requirements for assurance of quality and technical safety and assessment and verification of conformity of selected equipment [12], SÚJB Decree No. 408/2016 Coll., on management system requirements [13], and SÚJB Decree No. 329/2017 Coll., on basic design criteria for a nuclear installation [14].

Other licensing documents are related to the legislative pyramid, e.g. the SÚJB Decisions on Operation of nuclear installations supplemented with conditions of such Decisions, which require, for example, periodic ageing and life assessment of the most important equipment including incorporation of these outcomes in the annual update of the Final Safety Analysis Report as well as the date for periodic safety review. The last Decisions on Operation for the Dukovany NPP contains a condition for regular updating of the documents demonstrating the possibility of safe "long term operation" of the Dukovany NPP.

2.2 International standards

2.2.1 Dukovany and Temelín Nuclear Power Plants

As stated in Chapter 2.1, the WENRA Safety Reference Levels [3] (Issue I, as well as other associated safety reference levels, e.g. C, J, K, P, Q) have been implemented in Czech atomic law and also in Regulatory Safety Guide BN-JB-2.1 [5], which includes recommendations of IAEA Safety Guide No. NS-G-2.12 "Ageing Management for Nuclear Power Plants"[6].

The overall Ageing Management Programme in ČEZ, a. s. (i.e. holder of the license to operate both nuclear power plants) is set for both sites and includes requirements of the relevant IAEA Standards, IAEA Safety Guides (including newly developed SSG-48 "Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants") and WENRA Safety Reference Levels [3].

In the management system of ČEZ, a. s., the relevant WENRA Safety Reference Levels are reflected in the standard defining the basic rules for assuring nuclear safety in operation of a nuclear power plant, including basic rules for ageing management: ČEZ_ST_0065 Nuclear Safety in NPP Operation [15].

Other international documents used to develop the Overall Ageing Management Programme are as follows:

- IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles: Safety Fundamentals, Vienna 2006 [16]
- IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Specific Safety Requirements, Vienna, 2016 [18]
- IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), Safety of Nuclear Power Plants: Commissioning and Operation Specific Safety Requirements, Vienna, 2016 [19]

- IAEA Safety Standards Series No. SSG-25, Periodic safety review for nuclear power plants: specific safety guide, Vienna 2012 [17]
- IAEA Safety Reports Series No. 57, Safe Long Term Operation of Nuclear Power Plants, Vienna, 2008 [7]
- IAEA Safety Reports Series No. 82, Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL), Vienna, 2015 [22]
- IAEA Services Series No. 26, Guidelines for Peer Review of Safety Aspects of Long Term Operation of Nuclear Power Plants, Vienna, January 2014 [20]
- IAEA-TECDOC-1736, Approaches to ageing management for nuclear power plants: International Generic Ageing Lessons Learned (IGALL) Final report, Vienna, 2014 [21]

Regulatory Safety Guide BN-JB-2.1 is particularly reflected in the following processes and related management systemdocuments of ČEZ, a. s.:

- ČEZ_PG_0001 Ageing Management Programme for NPPs [37] (Overall AMP)
- ČEZ_PP_0404 Ageing Management for NPPs [23]
- SKČ_PP_0133 Assets Management Strategy [24]
- ČEZ_PP_0413 NPP Configuration Management and Design Basis Administration [25]
- ČEZ_ME_0987 Selection and Assessment of Equipement for AM and LTO [26]
- ČEZ_ME_0865 Development of the Component Specific Ageing Management Programme [27]
- ČEZ_ME_0870 Development of the Specific Ageing Management Programme [28]
- ČEZ_ME_1031 Determination and Development of TLAAs [29]

International standards and recommendations in the area of ageing management are therefore reflected in the overall AMP through the fulfilment of the requirements set out in the Czech legislative pyramid.

2.2.2 LVR-15 research reactor

The Ageing Management Programme for the LVR-15 research reactor has been developed on the basis of the requirements and recommendations referred to in IAEA Specific Safety Guide No. SSG-10: Ageing Management for Research Reactors [30] and IAEA-TECDOC-792: Management of research reactor ageing [31].

Other international standards and recommendations shall be reflected in the AMP for the LVR-15 research reactor through the fulfilment of the requirements set out in new legislation within the time limit defined by transitional provision of the "new" Atomic Act [2].

2.3 Description of the overall Ageing Management Programme

2.3.1 Scope of the overall Ageing Management Programme

2.3.1.1 Scope of the overall AMP for Dukovany and Temelín Nuclear Power Plants

Before the new Atomic Act entried into force, the ageing management process was implemented as an integral part of reliability management process in the management area of Assets management and fulfilled the requirements referred to in Regulatory Safety Guide BN-JB-2.1 [5].

Now, the separate ageing management process for NPP was developed to fulfill the requierements set out in the Atomic Act.

The overall AMP in ČEZ, a. s. is now set and implemented in accordance with SÚJB Decree No. 21/2017 Coll. [4]. In the integrated management system of ČEZ, a. s., the Ageing Management for NPP is a separate process within the scope of management area Engineering management and related to the management areas Assets management and Safety management. Within these management areas, the fundamental principles and strategies for ageing management are set and the responsibilities, obligations and powers are defined in the implementation of the activities carried out under that process. The objective of ageing management is to prevent failures of SSCs and ensure its long-term reliability and performance of safety functions. All areas of the management system of ČEZ, a. s. are documented by means of the management system documentation, the general structure of which is shown in Fig. 2.2.

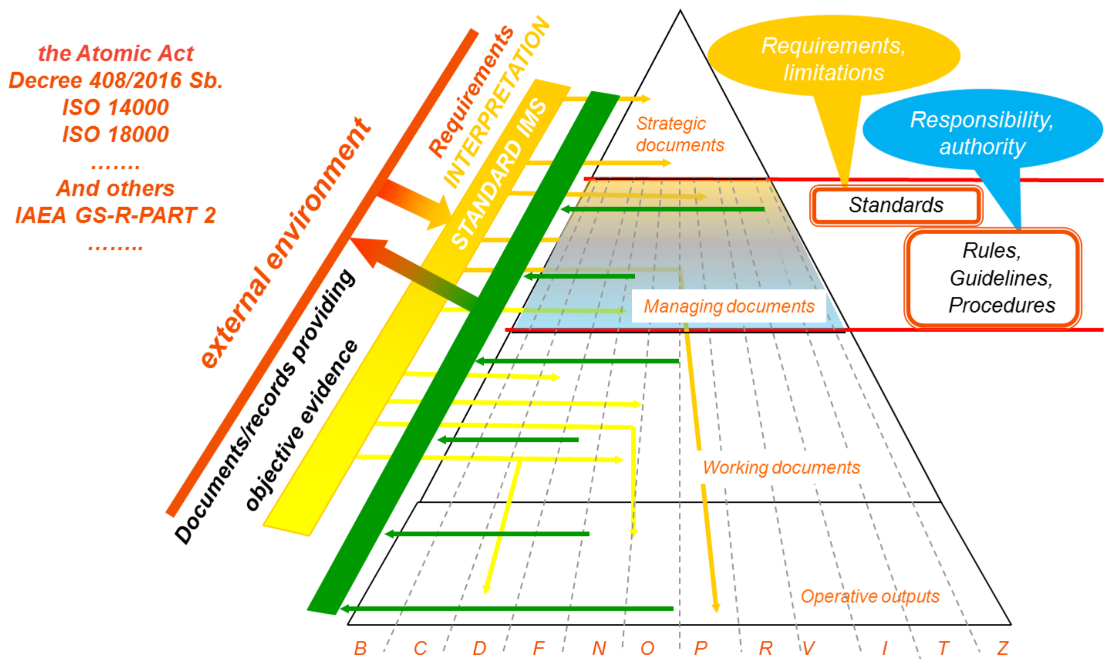

Fig. 2.2: General structure of the management system documentation of ČEZ, a. s.

Scheme of the Ageing Management for NPPs is shown in Fig. A.4 in Annex A.

In the ČEZ, a. s. integrated management system, the fundamental principles and the approach to ageing management are defined in ČEZ_ST_0006 Life management of equipement in Power Plants [32], ČEZ_ST_0072 Reliability Management Requirements [33][33] as well as ČEZ_ST_0065 Nuclear Safety in NPP Operation [15].

As stated in Chapter 2.2.1, the requirements for nuclear safety to be taken into account in operation of a nuclear installation – and therefore also the WENRA Safety Reference Levels, are implemented in ČEZ_ST_0065 Nuclear Safety in NPP Operation [15]. For the area of ageing management, the principles are as follows:

- An effective Ageing Management Programme is implemented for systems, structures, and components (SSCs) important to safety so as to continuously perform required safety functions throughout the service life of the plant, to continuously determine possible consequences of their defects , and to determine necessary activities in order to maintain the operability and reliability of these SSCs.
- The Ageing Management Programme shall be coordinated in accordance with other relevant programmes, e.g. the PSR. A systematic approach is applied to developing, implementing and continuously improving the Ageing Management Programme.
- Long-term effects of environmental conditions (temperature, corrosion effects and other degradations, which could influence long-term reliability of SSCs) as well as mode of operation (number of cycles, maintenance, testing) shall be evaluated and assessed as part of that Programme. The Programme shall take account of safety relevance of each SSC.
- The holder of license shall provide monitoring, testing, sampling and inspection activities to assess ageing effects and to identify unexpected behaviour and degradation during service life of all SSCs.
- The Ageing Management Programme shall be reviewed and updated as a minimum within the PSR process, in order to incorporate new information as it becomes available, to address new objectives and to assess the results of maintenance achieved. In addition, the PSR shall be used to confirm whether ageing and wear-out effects have been correctly taken into account and unexpected behaviours have been detected in a timely manner.
- In the design, margins shall be provided for all SSCs important to safety including relevant ageing, wear-out and age related degradation in order to ensure the capability of structures, systems and components to perform the necessary safety functions throughout their life. Ageing effects shall be covered by margins in the monitoring and testing, and shall be taken into account in all unit operating modes including PIE.
- Ageing management of the reactor pressure vessel shall include the effects of embrittlement, thermal ageing, and material fatigue and test results shall be compared with predictions, throughout plant life.
- Maintenance of major components (e.g. RPV and SG) shall be carried out to timely detect the ageing effects and to allow for preventive and remedial actions.

According to ČEZ_ST_0006 Life management of equipement in Power Plants [32], lifetime shall be managed for all equipment, in a graded manner in relation to the specified category of equipment. In order to ensure the required life of SSCs, graded ageing management of SSCs shall be implemented, ensuring the following requirements:

- Allowing timely detection of causes of ageing and mitigation of ageing effects of SSCs important to plant safety and operation, thus ensuring their long-term reliability
- Documenting to the regulatory body the maintenance of safety margins and residual life of SSCs, which are included in the scope of SSCs for ageing management
- Optimising the preventive maintenance programme in order to support ageing management particularly for critical equipment

- Allowing to determine a schedule for replacement/modernization programme of SSCs, which are no longer found suitable for operation from the point of view of safety, economic and, where appropriate, any other serious reasons in order to ensure the fulfilment of plant specification
- Allowing to extend the operation of SSCs beyond the original design life while ensuring the safe operation of the plant (the LTO Programme)
- Providing the basis for optimal utilisation of plant life

Strategies, principles and the desired objectives of ageing management are defined in ČEZ_ST_0072 Reliability Management Requirements [33].

The strategy for ageing management in the reliability management involves:

- Assessing each deviation from normal conditions in relation to possible ageing
- Reducing constant and operational load factors, thus mitigating the ageing of SC (prioritising predictive maintenance)
- Predicting trends in ageing in order to prevent unexpected failures (minimise failures for critical SC)
- Using specific and local indicators for detecting, monitoring and trending early stages of degradation (ageing) of SC
- Planning maintenance activities taking into account the current and predicted condition of SC

In order to ensure long-term reliability of SSCs and to ensure long-term operation of NPP:

- Activities (AMP) should be implemented to address the ageing management of SCs important to safety and production to ensure that the required functions important to safety and production are maintained throughout the lifetime of the plant; possible consequences of failure should be identified and the necessary actions should be defined to reduce degradation; operability and reliability of those SCs should be maintained including designation of responsible persons, departments, organisations and specification of the dates for implementation
- Current long-term plans should be developed and maintained for any more significant maintenance activities and modifications taking into consideration characteristics of passive and active components, the effect of ageing and obsolescence of equipment with regard to basic inputs and documentation
- Manufacturers'/suppliers' experience with regard to long-term operation of SC should be requested and used/implemented

The requirements of the aforesaid standards are reflected in overall AMP ČEZ_PG_0001 Operational Ageing Management Programme [37] and ČEZ_PP_0404 Ageing Management for NPPs [23], which are the managing procedures defining activities in the Ageing Management for NPPs, responsibilities, inputs and outputs of the process.

Assignment of responsibilities for the overall AMP:

Manager of NPP Design Authority department, who is the guarantor of the Ageing Management for NPPs, is responsible for setting the process indicators and controlling compliance with the procedure, implementing remedial actions in order to continually improve the process and has right to request cooperation of the employees concerned, who carry out the activities in the Ageing Management for NPPs.

Responsibility for implementing ageing management is assigned to those employees performing particularly the roles of the ageing management specialists for NPPs (NPP Long-term Operation Preparation department), system engineer, component engineer (Care of Assets department), and segment engineer (NPP Engineering department). Assignment of activities (responsibilities) to the individual roles is described in ČEZ_PP_0404 Ageing Management for NPPs [23].

Methods used for identifying SSCs within the scope of overall AMP

The requirement for scope setting of systems, structures and components subject to the ageing management process is generally defined in SÚJB Decree No. 21/2017 Coll. [4]. The following should be included in the selection of systems, structures and components subject to the ageing management process:

- Selected equipment (safety classified); and
- Safety related SSCs, which are not the selected equipment.

In addition, according to the requirements of the SÚJB Decree No. 162/2017 Coll. [9], the results of the probabilistic safety assessment shall be used to verify the scope of SSCs subjected to the ageing management process.

In ČEZ_ME_0987 Selection and Assessment of Equipement for AM and LTO [26], criteria for scope setting of equipment subjected to the ageing management are set out.

Identification of equipment falling into the scope of AM is based on the entirelist of all equipment registered in the plant's equipment register (the EAM Asset Suite system is now being used). From the list of all NPP equipment the following equipment is selected for the purposes of AM:

a) All selected equipment under the Atomic Act [2] (equipment with the assigned Safety Class 1, 2, 3)
b) Equipment with the criticality level 1 and 2 assigned under ČEZ_ME_0608 [34] and equipment fulfilling the safety function of category 1 or 2 important to nuclear safety (under ČEZ_ME_0901 [35])
c) Equipment recommended from the PSA
d) Other equipment recommended on the basis of global good practice and operating experience

According to ČEZ_ME_0608 [34], relevance of all functions shall be specifically identified for each technological system (TS), in terms of the impact on performance of the safety functions, safe shutdown and energy production. In the next step, all equipment in TS are assessed from a perspective of impact of their failure on performance of the defined technological functions of the system and included in the relevant criticality category.

The related methodology ČEZ_ME_0901 [35] defines which TS and subsequently which SCs in identified TS are important in terms of performance of the safety functions and therefore how the SCs are relevant to safety. This methodology classifies SSCs in terms of impact on nuclear safety (in terms of SSCs relevance in management of consequences of postulated initiating event).

The assigned safety relevance to individual items important to nuclear safety is the outcome of this classification. SSCs performing operational functions, whose failure does not result in exceeding of the parameters above the values specified in Limits and Conditions of Safe Operation (LaC), are classified as irrelevant to safety.

Criticality tables created within the Effective Maintenance Strategy project serve as a basis for selecting (screening) SCs according to the above criterion referred to in this chapter.

Grouping methods of SSCs in the scope of the overall AMP

Grouping equipment into commodity groups is possible according to BN-JB-2.1 - Annex 2, point 5 [5] and is in conformity with the global good practice applied in order to maximize work efficiency. The grouping of equipment shall be carried out according to the methodology ČEZ_ME_0987 Screening and Assessment of Equipment for LTO [26]:

Grouping is carried out on the basis of the following features:

a) Identical maintenance template
b) Commodity classification (valve bodies of one type series, pump casings, tanks of similar technical type, pressure vessels of similar technical type, etc.)
c) Identical identified degradation mechanisms/ageing effects, which represents subsequently grouping by:
    - Mode of operation
    - Physical parameters of the medium
    - Chemical composition of the medium
    - And, if appropriate, other nuances of operation, if any

Methodology and requirements for evaluation of the existing maintenance practices and developing of new AMPs appropriate for the identified significant degradation mechanism

The Ageing Management Review (AMR) is used to assess the existing maintenance activities and develop new AMPs. The verification of ageing effect management, i.e. whether the degradation mechanisms and ageing effects identified are properly managed, involves the assessment whether:

- The existing methods of monitoring, detection, prediction, evaluation and mitigation of ageing of the equipment are sufficient for management of the identified significant effects of ageing and degradation mechanisms;
- Timely detection and mitigation of the effects of ageing mechanisms are provided by existing specific ageing management programmes for NPP equipment. These group of programmes also include other programmes like programmes for operation, diagnostics, testing, inspections and maintenance, that have been established since the start of operation with the same objectives of which are described under previous bullet.

The assessment shall be always carried out from two perspectives:

- Recommendation to implement the AMP based on global general experience;
- On the basis of an analysis of the current state of maintenance (care of equipment); i. e. on the basis of information from real operation - whether an appropriate AMP to be implemented is assigned to each identified degradation mechanism/ageing effect.

For Dukovany NPP, this review was updated recently during the preparation for long term operation (LTO).

For Temelín NPP, the AMR is being updated between 2016 and 2018 in relation to the upcoming PSR for the Temelín NPP after 20 years of operation.

At the same time, evaluation of the maintenance practices was carried out on both sites between 2011 and 2014 under the Effective maintenance strategy project (according to ČEZ_ME_0898 Effective Maintenance Strategy [36]), which identified failure modes for individual design types of equipment with the use of the EPRI PMBD and operating experience; for those failure modes, new maintenance practices were subsequently defined depending on the criticality of each individual equipment. Results of that assessment are one of the inputs for the AMRs.

Quality assurance of the overall AMP (in particular, collection and storage of data and trending of information on maintenance history and operational data, indicators used to assess the effectiveness of the process)

Quality assurance for the ageing management process required in the SÚJB Decree No. 21/2017 Coll. [4] is defined and described in ČEZ_PG_0001 Operational Ageing Management Programme for NPPs [37].

The required activities in the process of ageing management defined in the Ageing Management Programme are described in ČEZ_PP_0404 [23]; the effectiveness of that process is evaluated through the AMR. This process supports the assets management process, as described in SKČ_PP_0133 [24], the effectiveness of which is evaluated for individual technological systems through Health Reports, as described in ČEZ_ME_0919 [56], and which is used to monitor the performance and state of SSCs based on the monitoring of a set of TS parameters for the specified areas for assessment. The purpose is to receive feedback on the current performance, state of TS and its SCs, and the effectiveness of the maintenance programmes, and to timely identify the signs of unfavourable development in performance and state for the TSs to be assessed, in order to optimise the maintenance strategy and measures to fulfil the required level of performance and technical-economic specification of the NPP; in addition, the relevant outputs are part of the assessment for the purposes of documenting preparedness to ensure long term operation (LTO) of the plant.

The evaluation of the Health Reports includes but is not limited to the following parameters:

- Unplanned entry into operational limits and conditions (LaC)
- Planned entry into operational limits and conditions for repair of equipment
- Non-compliance with LaC
- Number of operational events
- Corrective maintenance statistics by urgency
- Trend in the number of failures
- Trends in the costs of preventive and corrective maintenance
- Outputs of the assessment of ageing management

- Loss of production due to the particular technological system
- Unit power reduction due to the particular technological system
- Unit shutdown due to the particular technological system

The key performance indicator “Monitoring of the effectiveness of the ageing management process for safety relevant equipment” is evaluated at the level of the whole plant, i.e. proportion of the number of malfunctions or failures/power reduction due to ageing to the overall number of malfunctions or failures/power reduction, arising from the absence or poor setting of the requirement or criterion under the Ageing Management Programmes. In addition the percentage of the remedial actions implemented to the total number of recommendations from the process of ageing management is evaluated. At the same time, the effectiveness of the individual specific AMPs is monitored depending on the nature of these AMPs and their parameters being monitored.

In case of non-conformities related to equipment ageing, these shall be assessed and remedial action shall be proposed (in accordance with ČEZ_PP_0404 [23] and SKČ_PP_0133 [24]). The effectiveness of remedial actions is assessed in the Health Reports.

2.3.1.2 Scope of the overall AMP for the LVR-15 research reactor

At the end of 2016, the Czech legislative environment did not contain an explicit term the Ageing Management Programme. That does not mean, however, that previous legislation did not contain requirements to monitor physical condition of the SSCs important to safety, to carry out maintenance, in-service inspections and for selected equipment included in Safety Class 1 or 2 (note: LVR-15 has no equipment included in Safety Class 1), requirements to define the criteria for life monitoring of such selected equipment. These requirements are, with the use of the principle of graded approach, detailed in Regulatory Safety Guide BN-JB-1.15 [8], which in addition to the detailing of the aforesaid requirements, provides specific recommendations concerning the area of ageing of nuclear research facilities.

As stated in Chapter 2.1, the terms “ageing management process” and “ageing management programme”, under which the process should be carried out, are introduced in the new Atomic Act, and the details of that programme are also specified.

The Ageing Management Programme for the LVR-15 research reactor will be brought into line with the new legislative requirements within the time limit defined in the transitional provisions of the “new” Atomic Act by the end of 2018.

Assignment of responsibilities for the AMP

The Ageing Management Programme for the LVR-15 research reactor falls under the responsibility of Reactor Operation Director of the company Centrum výzkumu Řež s.r.o. (Research Centre Řež).

Methods used for identifying SSCs within the scope of AMP for the LVR-15 research reactor

In the Ageing Management Programme for the LVR-15 research reactor, reactor systems, structures and components (SSCs) the ageing of which should be monitored are identified, in conformity with the above mentioned IAEA documents [30] and [31].

In principle, this is selected equipment, which has an impact on the nuclear safety and which has an impact on the operational reliability of reactor. The Programme does not cover the experimental facilities used in that reactor, second and third cooling circuits, and dosimetry system.

The SSCs were included in the AMP on the basis of an analysis of degradation mechanisms and ageing effects on individual selected reactor equipment and the degree of their impact on the nuclear safety in accordance with the rules set out in the IAEA documents [30] and [31].

Methodology and requirements for evaluation of the existing maintenance practices and developing of new AMPs appropriate for the identified significant degradation mechanism

Evaluation of the existing maintenance practices is based on strict control of the individual inspections carried out under the in-service inspection programme for selected equipment, according to which the maintenance of equipment is carried out. In addition, such evaluation is based on keeping of individual reports, comparing the results of inspections and where any change is detected in the parameter being monitored, investigating causes. An independent evaluation of the practices carried out is conducted in accordance with the internal guideline OSM 29 (Nuclear Safety Assurance) in reactor operation by an independent committee, which assesses the nature and results of operations at regular intervals, and also the system of internal audits of operations.

Quality assurance of the AMP for LVR-15 research reactor (collection and storage of data and trending of information on maintenance history and operational data, indicators used to assess the effectiveness of the process)

The ageing management process was not explicitly defined by law until 2017 and therefore, the existing ageing management process itself has not been incorporated as a separate process in the reactor quality system. Quality assurance for data collection, storage of records and assessment system for the area associated with ageing management is based on the operational quality system / reactor control system in the area of processes for the planning and control of in-service inspections and maintenance and repairs of selected equipment.

The Ageing Management Programme was updated in the light of the results of regular and special inspections, scientific and technological developments in the area of detection and new means of detection, events in similar installations in the world, and internal and external operating experience. In the context of compliance with the requirements set out in new legislation for the deadline defined by transitional provision of the "new" Atomic Act [2], that Ageing Management Programme will be harmonised.

All records of inspections and other activities are documented in the designated place with reactor engineer and in electronic form in the place intended for the storage of records of reactor maintenance.

2.3.2 Ageing assessment of systems, structures and components

2.3.2.1 Ageing assessment of systems, structures and components of the Dukovany and Temelín Nuclear Power Plants

As mentioned above, the ageing management process for Dukovany and Temelin NPPs is set on the basis of the so-called "Ageing Management Review", in which all significant potential and real degradation mechanisms and ageing effects have been identified for all systems, structures and components relevant to safety. The ageing management itself is ensured by applying a graded approach, according to ČEZ_ST_0006 [32], specifically:

a) With the use of specific and component specific AMPs;

b) With the use of standard methods of preventive maintenance in the framework of performance and condition monitoring.

The effectiveness of the method of ageing management chosen is monitored at the level of both individual Programmes and the overall AMP.

Use of key standards, guidance and manufacturing documents in the preparation of the overall AMP

Standards, guidance and manufacturing documents are used in several parts of the overall AMP:

- In the area governing the development of component specific AMPs (ČEZ_ME_0865 Development of the Component Specific Ageing Management Programme [27]) and preventive maintenance setting (ČEZ_ME_0225 Preventive Maintenance in Asset Suite for NPP [38]), the manufacturer's recommendations referred to in the accompanying technical documentation of equipment and the legislative requirements also serve as a basis.
- In the area governing the process of ageing management review, specifically at the stage of understanding of ageing.

Key elements used in plant programmes to assess ageing

The following programmes are considered as other plant programmes important to ageing management, in conformity with IAEA SRS No.57 [7]:

1. In-service Inspections Programme
2. Maintenance Programme
3. Programmes for monitoring and control of operating modes including inspection activities in the framework of operation, pressure and leak tests, inspection activities defined in the LaC and surveillance specimen programmes.
4. Chemistry Control Programme
5. Equipment Qualification Programme
6. Operating staff walk-downs

Plant programmes, important to ageing management, are assessed, in conformity with the national and international requirements, in terms of required characteristics of the effective Ageing Management Programme (nine attributes), i.e. fulfilment of the following areas was assessed:

1. The scope of the Programme is defined
2. Preventive actions, activities to mitigate the effects of ageing and to control the effects of ageing are defined; the controlled parameters are defined
3. Methods and means of monitoring degradation mechanisms and ageing effects are defined
4. Monitoring and trending of the parameters to be monitored are implemented
5. Acceptance criteria are established
6. Remedial actions are defined
7. Process of confirmation of the carried out activities is implemented

8. Management system is implemented
9. System for operating experience feedback is implemented

Processes and procedures for the identification of degradation mechanisms and their possible consequences

The method for the identification of degradation mechanisms is described in ČEZ_ME_0987 Scoping and Assessment of Equipment for AM and LTO [26]:

Understanding of ageing is a key prerequisite for effective monitoring of the course of ageing and for mitigation of the effects of equipment ageing.

Significant ageing effects are identified for each commodity group, specifically:

- Assumed (potential) based on global general experience on the basis of the assessor's knowledge of equipment and on the basis of the mode of operation (based on the catalogue of degradation mechanisms of the Dukovany and Temelín Nuclear Power Plants [39]).
- Identified (real) on the basis of operator´s experience and on the basis of real operation of NPP

The ageing effects assumed (potential) based on global general experience are identified on the basis of an analysis of the following documents:

- Generic Ageing Lessons Learned (GALL) Report NUREG 1801, Rev2 [40]
- Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned, SRS No. 82 (IGALL), 2015 [22]
- IAEA-TECDOC-1025, Assessment and management of ageing of major nuclear power plant components important to safety: Concrete containment buildings, 1998 [41]
- ACI 349.3R-02, Evaluation of Existing Nuclear Safety-Related Concrete Structures, Ronald J. Janowiak a spol, 2002 [42]
- EPRI documents (http://www.epri.com/):
    - Nuclear Maintenance Applications Center: Passive Components Maintenance Guide for Nuclear Power Plant Personnel [43]
    - EPRI Technical Report, Augmented Containment Inspection and Monitoring Report, 2013 [44]
    - Information from the PMDB (Preventive Maintenance Database)
- The ageing effects assumed due to operation are identified on the basis of such information of the design and other documents of SSCs:
    - Material of which equipment is made and material properties
    - Type of steel, in particular in the basic subdivision into stainless (austenitic steel) x carbon steel
    - In specific cases, material could be identified according to ČSN (Czech national standards) and if appropriate, other applicable standards
- Manufacturing methods (forged piece, cast piece, other …)
- Mode of operation (stand by/on line)
- Temperature, pressure and other properties of media, as necessary
    - Flow velocity
    - Stratification

- Chemical composition of the media (concentration of oxygen, hydrogen, hydrazine, chlorine and if appropriate, other corrosion-active substances)
- General data resulting from research and operating experience

Real ageing effects detected in operation are identified on the basis of information analysis from the following areas:

- History of operation, inspections and maintenance; the main sources of information are particularly:
  - Health reports
  - Work orders
  - Work requests
  - Records in the SIS (events database)
  - Database of unplanned entries into operational limits and conditions
  - Meeting minutes of the Commission of failure investigation
  - Condition Assessment Report for unit startup
  - If necessary, consultations with system and component engineers
- Results of the examination of equipment taken out of service
- Functional reliability assessments and equipment lifetime evaluations , carried out at the time from the start of operation

Outputs of the research project of the ČEZ company and the Ministry of Industry and Trade (Effective Long-term Operation of Nuclear Power Plants in the Czech Republic) are also used for identification of degradation mechanisms. Under this project, a catalogue of degradation mechanisms of Czech NPPs [39] and documents describing typical degradation mechanisms and methods of their management were compiled for the following typical equipment:

- Motor Operated Valve [45]
- Air Operated Valve [46]
- Pipe section [47]
- Flange joint [48]
- Pressure vessel [49]
- Heat Exchanger [50]

Methods of establishing acceptance criteria

Parameters and their limit values for ageing management are defined in the process ČEZ_PP_0413 Configuration Management and Design Basis Administration [25], which provides them in relation to the equipment design, operational requirements, internal and external operating experience. That information is provided as Equipment Reliability Management (ERM) parameters.

In order to determine parameters and their limit values for ageing management, the following sources are particularly used:

- Design basis documents (at the level of technological system as well as at the level of individual SCs), in particular parameters for the SSCs margin management
- LTO documentation (AMR, TLAA, , Specific comprehensive assessment reports)
- SSCs technical documentation – in particular technical specifications, IQAPs, etc.
- Operating experience and other calculations and analyses

In the event that the parameters and their limit values cannot be determined with the use of the above mentioned sources, an analysis of understanding of ageing should be carried out and parameters and their limit values should be determined in that analysis.

Use of R&D programmes

Research and development programmes are used in the course of the AMR for identifying degradation mechanisms and ageing effects (see paragraph "Processes and procedures for the identification of degradation mechanisms and their possible consequences"). In addition, they were used for developing the overall Ageing Management Programme. At the very beginning of programme setting (2003 - 2007) the research project of the Ministry of Industry and Trade and ČEZ, a. s., aiming at developing and verifying the procedures for the ageing management of NPPs, was implemented. Basic methodologies for ageing assessment and management, catalogue of degradation mechanisms, database application for supporting life and ageing management, etc., have been developed under that project.

In addition, national, international and company R&D programmes are used for the Ageing Management Programme. The company ČEZ, a. s., is a member of the EPRI (Electric Power Research Institute), where the company is engaged in R&D, inter alia in the following programs:

- Materials Degradation / Ageing
- Fuel Reliability Program
- Used Fuel and High-Level Waste Management Program
- Non-destructive Evaluation Program
- Nuclear Maintenance Application Centre
- Long Term Operations Program

For example, the following R&D programmes have recently been implemented in the area of ageing management at the national (Ministry of Industry and Trade, Technology Agency of the Czech Republic etc.) and at the corporate level (ČEZ):

- Methodology for determination of the life of high-voltage insulation systems for rotating machines
- Development of the methodology "Life Specification of Fixed Type Transformer Insulation in Order to Eliminate Operational Risks"
- Development of the methodology for diagnostics of safety cables in operation in a moderate environment and non-safety cables
- Research and development of new diagnostic systems for life assessment of turbines
- Methods and tools for inspections and tests - Preparation of risk-oriented program of periodic tests on valves
- Development of the assessment method for environmentally assissted fatigue
- Ensuring long-term operation of NPP reactor internals

At the international level, this includes, in particular, the R&D programmes on the NUGENIA platform, of which both ČEZ and its subsidiary company ÚJV Řež, a.s. are members. It was cooperation, for example, in the following R&D projects in the area of ageing:

- DEFI PROSAFE - DEFInition of reference case studies for harmonized PRObabilistic evaluation of SAFEty margins in integrity assessment for long-term operation of reactor pressure vessel

- AGE 60+ - Applicability of ageing related data bases and methodologies for ensuring safe operation of LWR beyond 60 years
- SOTERIA - Safe long term operation of light water reactors based on improved understanding of radiation effects in nuclear structural materials
- MAPAID - Modelling and Application of Phased Array ultrasonic Inspection of Dissimilar metal welds
- INCEFA – PLUS - INcreasing Safety in NPPs by Covering gaps in Environmental Fatigue Assessment

The results of these programmes are used in the process of ageing management for ČEZ NPPs.

Other R&D programmes concerning ageing management for specific systems and components are referred to in Chapters 3.1.2.1. (Electrical Cables), 4.1.2.1 (Concealed Pipework) and 5.1.2.1 (RPV).

Use of internal and external experience

The method of use of internal and external experience at the stage of ageing management assessment is described in Chapter 2.3.2.1, section "Processes and procedures for the identification of degradation mechanisms and their possible consequences". The method of use of internal and external experience under the Ageing Management Programme is described in Chapter 2.4.1, section "Evaluation of plant specific experience, external experience, taking account of current state of art including R&D results".

2.3.2.2 Ageing assessment of systems, structures and components of the LVR-15 nuclear research reactor

Use of key standards, guidance and manufacturing documents in the preparation of the AMP

The Programme has been prepared using the requirements and recommendations referred to in [30] and [31]. Manufacturing documents of the major parts of reactor systems and components including IQAP and manufacturers' and suppliers' recommendations were used in the preparation of the AMP, on the basis of which the list and programme of in-service inspections for selected equipment have been developed, as well as supplementary technical reports in the area of condition assessment of LVR-15 research reactor components. The in-service inspection programme, independent life assessment of selected LVR-15 research reactor components and interim reports drawn up by the ÚJV Řež, a.s. and the National Research Institute for Materials are the key elements in the area of ageing assessment.

Elements in terms of an effective Ageing Management Programme, as stated in [30], include equipment scope-setting for ageing management; identification of degradation mechanisms/ageing effects (understanding of ageing); minimisation of ageing effects; detection, monitoring and mitigation of ageing effects; continuous improvement in the Programme, and record keeping.

Processes and procedures for the identification of degradation mechanisms and their possible consequences

Inspections under the In-service Inspection and Equipment Maintenance Programme are the key process for the detection of ageing effects. Other actions include internal and external feedback information, results of the research projects, recommendations, external analyses, calculations, walkdown activities and other special inspections.

Methods of establishing acceptance criteria

The acceptance criteria for ageing of SSCs are established on the basis of safety analyses and the associated operational limits and conditions and, if appropriate, on the basis of required functions of the SSCs concerned as well as according to the established acceptance criteria for individual tests in the course of inspections on selected equipment and, pre-operational and operational tests for the demonstration of the fulfilment of limiting conditions. Critical components such as reactor vessel, internals and heat exchangers were also independently assessed including recommendations in the area of inspections, and design and technological modifications. In the area of critical components such as the reactor vessel itself, the overall neutron fluence and the assessment of potential for radiation damage having regard to the expected neutron fluence in operation are the established and assessed criteria. In addition, the review shall assess the potential for achieving limit parameters for structural materials of major components.

Use of R&D programmes

The condition assessment of vessel and internals shall include the determination of the fluence achieved in relation to such materials. That determination was made using the calculations with the validated programme on the basis of measurement with activation detectors. To evaluate the working conditions under the vessel head, gamma radiation dose rates and neutron measurements were carried out in that area. In the context of the conversion of IRT2M fuel to IRT4M fuel in relation to a change in enrichment, research reports addressed changes in neutron flux in the vessel including changes in spectra and impact on radiation conditions inside the reactor.

To assess degradation mechanisms for structural materials, long-term experience is used in the field of R&D and testing within the ÚJV Řež, a.s. company under the surveillance programme and irradiated material testing in hot cells – knowledge and experience are subsequently used in the independent condition assessment of reactor structural materials.

Use of internal and external experience

The Ageing Management Programme shall be regularly updated in the light of operating experience, results of the inspections and tests of individual equipment. The Programme includes independent reports, using the results of international experience in the area of life assessment and management such as IAEA-TECDOC-792 [31] (Ageing Problems Reported in Research Reactor), or in the area of assessment and assessment methods such as IAEA Consultants meeting "Assessment of Core Structural Materials and Surveillance Programme of Research Reactors", IAEA-TECDOC-1263 "Application of non-destructive testing and in-service inspection to research reactors". In some inspections, such experience and recommendations led to a change in inspection procedure, adjustments to the time schedules, extension of the list of the equipment tested as well as method of inspections – individual changes are recorded in the framework of the relevant revisions of the inspection programme for selected LVR-15 research reactor equipment.

2.3.3 Monitoring, testing, sampling and inspection activities

2.3.3.1 Monitoring, testing, sampling and inspection activities in Dukovany and Temelín Nuclear Power Plants

Programmes for monitoring condition indicators and parameters and trending

Continuous monitoring of defined parameters and their trends, comparison of the identified data with the acceptance criteria, and identification of deviations are part of the continuous monitoring of performance and conditions of SSCs under the Assets Management Strategy [24]. Continuous performance and condition monitoring includes:

- Continuous monitoring of defined parameters and their trends, comparison of the identified data with the criteria, and identification of deviations
- Assessment of the results of the AMPs (component specific, specific)
- Assessment of the findings of walkdowns
- Assessment of the results of field tests, inspections, revisions, diagnostics
- Assessment of the results of chemistry evaluation
- Assessment of the documentation of maintenance (non-conforming reports of in-service inspections and testing)
- Monitoring of the validity of qualified life
- Monitoring of the fulfilment of qualification requirements
- Assessment of deviations, non-conformities, failures and requirements including near miss

The Ageing Management Programmes are the main programmes for monitoring equipment condition, and monitoring and assessing its development trend. There are two types of AMPs used in Dukovany and Temelín NPPs: Component Specific Ageing Management Programmes (described in the documents marked ČEZ_TST_XXXX) and Specific Ageing Management Programmes (described in the documents marked ČEZ_ME_XXXX). The Ageing Management Programmes have been developed and possibly adjusted in the light of the results of the ageing management review. The Ageing Management Programmes in Czech NPPs are listed in Table 2.1:

Document title:	Document ID code:
AMP for Main Circulation Pump	ČEZ_TST_0004
AMP for Pressurizer	ČEZ_TST_0006
AMP for Containments and Hermetic Areas of the Dukovany NPP and the Temelín NPP	ČEZ_TST_0014
AMP for Steam Generator	ČEZ_TST_0015
AMP for Main Isolation Valve	ČEZ_TST_0021 (only EDU)
AMP for Pipelines and Sections of Safety Class 1	ČEZ_TST_0023
AMP for Safety Relevant Cables in NPP	ČEZ_TST_0024
AMP for Cooling Towers	ČEZ_TST_0025
AMP for Fuel Storage and Refuelling Pools	ČEZ_TST_0031
AMP for Reactor	ČEZ_TST_0033
AMP for Boundary Valves of Safety Class 1in NPP	ČEZ_TST_0034
AMP for Oil-type Power Transformer	ČEZ_TST_0035
AMP for Generators	ČEZ_TST_0036
AMP for Turbines	ČEZ_TST_0040
AMP for High Energy Lines of ČEZ NPPs	ČEZ_TST_0054

AMP for NPP Reactor Pressure Vessels	ČEZ_ME_0780
AMP for Low-Cycle Fatigue - Passive Mechanical Components	ČEZ_ME_0773
AMP for Erosion Corrosion - NPP Secondary Circuit Piping	ČEZ_ME_0778
AMP for Safety Cables in NPP	ČEZ_ME_0791
AMP for Visual Inspections - NPP Cables	ČEZ_ME_0941
AMP for Fuel Storage and Refuelling Pools in Dukovany NPP	ČEZ_ME_0936
AMP for Containments in the Dukovany NPP	ČEZ_ME_0937
AMP for Civil Structure Parts of Pools with Double Liner in the Temelín NPP	ČEZ_ME_0964
AMP for Civil Structure Parts of Containment in the Temelín NPP	ČEZ_ME_0966
AMP for Pre-construction Condition Surveys of buildings	ČEZ_ME_0940
AMP for Measuring of Civil Structures Components	ČEZ_ME_0934
AMP for Monitoring of Dukovany NPP buildings	ČEZ_ME_1029
AMP for AMP for Monitoring of Structures of the Dukovany NPP	ČEZ_ME_1030
AMP for Material Diagnostics of Steam Turbines	ČEZ_ME_0990
AMP for Vibrational Diagnostics of Rotating Equipment	ČEZ_ME_0972
AMP for Dynamic Testing of TG Foundations	ČEZ_ME_0989
AMP for Inspection of the Deformations of Foundation-TG System	ČEZ_ME_0988
AMP for Electric Parameters Diagnostics of HV Rotating Electric Machines	ČEZ_ME_0986
AMP for Visual Inspections of HV Rotating Electric Machines	ČEZ_ME_0968
AMP for Diagnostics of Ozone Detection in Coolant of HV Rotating Electric Machines	ČEZ_ME_0970
AMP for Noise Diagnostics of HV Rotating Electric Machines	ČEZ_ME_0971
AMP for Alternator Rotor Rings	ČEZ_ME_0991
AMP for Stator Magnetisation Test of HV Rotating Electric Machines	ČEZ_ME_0979
AMP for Oil-type Power Transformer – Dissolved Gas Analysis (DGA)	ČEZ_ME_0985
AMP for Oil-type Power Transformer –Degradation of Solid Insulation in Transformer	ČEZ_ME_0984
AMP for Oil-type Power Transformer – Electrical Measurements on Winding Insulation System and Bushing with Measuring Lead	ČEZ_ME_0983
AMP for Oil-type Power Transformer – Diagnostics of Insulating and Qualitative Parameters of Insulating Oil	ČEZ_ME_0967
AMP for Oil-type Power Transformer – Solution of Impacts and Prevention of the Effects of Corrosive Sulphur in Oil Filling of Power Transformers	ČEZ_ME_0959
Ageing Management Methodology for Ionisation Chambers of PRPS – NIS Power Range Channels in the Temelín NPP	ČEZ_ME_0890
AMP - Ionisation Chambers of PRPS – NIS Power Range Channels in the Temelín NPP	ČEZ_ME_0908
AMP for Valves with Drives – Ageing Assessment Methods	ČEZ_ME_0999
AMP for Areas at Risk with Weld Joints in NPP	ČEZ_ME_0980
AMP for Concealed (Buried) Piping	ČEZ_ME_1036
AMP for Service Water Piping	ČEZ_ME_1043

Table 2.1: List of Ageing Management Programmes in Czech NPPs

Inspection programmes

The in-service inspection programmes for Dukovany NPP and Temelín NPP are put in place from the beginning of the operation of nuclear power plant. The inspection programmes were developed on the basis of Individual Quality Assurance Programmes (IQAP), prepared by manufacturers of individual equipment and if appropriate, by suppliers of such equipment, in the case of foreign supplies. The IQAP were approved by the former regulatory body – the Czechoslovak Commission for Atomic Energy. The Individual Operational Quality Assurance Programmes, prepared by the Dukovany NPP served as another source for developing the in-service inspection programmes for equipment which was later included in the list of selected equipment. That list was drawn up on the basis of Decrees of the Czechoslovak Commission for Atomic Energy No. 436/1990 Coll. Specifications for TS and SSCs operation and maintenance and if appropriate, instruction manuals were used as other sources.

The In-service Inspection Programme is a living document. Changes are caused by lessons learned from operation, in particular the results of inspections following the experience in various fields of testing. In addition, new inspection methods apply depending on their development and new potential for application. There are also changes in test methods in connection with higher testing sensitivity. There was also a change in the number and interval of pressure tests, the application of which causes higher utilisation of the residual life of equipment. Experience from other plants shall also apply in the in-service inspection programme. This is based on the exchange of information between experts of different professions, in particular findings of evaluation committees of the results of in-service inspections as well as lessons learned from the exchange of experience at specialised meetings organised by the IAEA in order to increase awareness among employees in the field of in-service inspections. WANO operator reports on the events which occurred in individual plants are another source of information. Such reports are analysed and if necessary, new points of inspection are included in the inspection programme, or different or new inspection methods are put in place.

The In-service Inspections Programmes are updated according to ČEZ_ME_0351 Development of Inspection Programmes and Plans, their Implementation and Assessment in NPP [51].

Surveillance programmes

These programmes shall include the activities carried out under the Monitoring and Control of Operating Modes. The Programme is implemented in accordance with Regulatory Safety Guide No. BN-JB-1.9 “Maintenance, In-service Inspections and Surveillance” [52], and IAEA Safety Guide No. NS-G-2.6 Maintenance, Surveillance, and In-service Inspection in Nuclear Power Plants [53]. Their implementation in NPPs is driven by Limits and Conditions of Safe Operation [54] and internal managing procedures for Monitoring and Control of Operating Modes [55]. Plant operation monitoring is based on the monitoring of the condition of technology, important operating parameters of the unit and LaC, discussion of potential remedial actions, evaluation of unit transients in case of abnormal or failure conditions. Furthermore, the following shall be carried out: independent evaluation of periodic and single tests and functional tests of unit technology; determination of the parameters and limits for the action of protection and limitation systems and alarm system in the relevant operating regulations; drawing up a time schedule for tests and protection and limitation systems. Independent control is carried out for compliance with the local operating procedures, operating instructions of operational programmes, Limits and Conditions, and valid control documentation by operating personnel. In addition, the Programme includes continuous inspection of the most important unit parameters in all operating modes and provision of feedback, transmission of primary information and knowledge on the condition and behaviour of technology, and evaluation of changes in important parameters to the workers of operating modes group for further processing. Programme documentation defines periodic inspections of unit operation; procedures for evaluation of unit shutdown and startup; failure management at reduced power level or unit shutdown by failure; equipment operability tests; documentation outputs including category, period of storage, and place of storage.

The following three programmes using representative material samples are currently implemented:

- Surveillance Specimen Programme for Reactor Pressure Vessel;
- Surveillance Specimen Programme for Hermetic Zone Concrete;
- Surveillance Specimen Programme for Safety Cables.

Provisions for identifying unexpected effects of degradation mechanisms

The effects of any unexpected degradation mechanisms shall be identified through the following activities

a) Walkdown activities of operating personnel.
b) If necessary, one time expert inspections (e.g. in the revalidation of TLAA for pump electric motors)
c) Indirect monitoring – monitoring of failures and findings abroad (e.g. sharing experience with the operator of VVER 440 in Slovakia), monitoring of conceptual ageing.

2.3.3.2 Monitoring testing, sampling and inspection activities - LVR-15 research reactor

Programmes for monitoring condition indicators and parameters and trending

Condition monitoring is based on the records of the individual inspections carried out under the in-service inspections plan; records of the complex testing carried out before each reactor startup in conformity with the Operational Limits and Conditions; keeping of the records on the changes made in technology; operational records on compliance with the Limits and Conditions of Safe Operation, thus fulfilment of the expected and required operating conditions of equipment; records to demonstrate performance of primary chemistry according to the LaC; records of possible investigations of non-conformities and causes and potential impacts including remedial and preventive actions; initial record on the state of reactor control and protection systems during operation; results of dosimetry measurements to demonstrate the functionality and integrity of the barriers against release of radioactive material.

In-service inspection programme

The In-service Inspection Programme for the LVR-15 research reactor in its current form has been in place since 1988, when the general reconstruction of the reactor was carried out. This involved replacing all process systems except for heat exchangers. These have been in operation since 1974. The In-service Inspection Programme is based on the Individual Quality Assurance Programmes (IQAP), provided by the suppliers of such equipment and approved by the former Czechoslovak Commission for Atomic Energy. In addition, the In-service Inspection Programme for the LVR-15 research reactor includes instructions and procedures for carrying out basic periodic maintenance (e.g. lubricating grease and oil replenishment, cleaning of switchgears, motors, etc.). Therefore, it was also based on the specifications for equipment operation and maintenance, on the instruction manual for equipment and operating experience of similar installations.

The In-service Inspection Programme is updated on a continuous basis in the light of operating experience, results of inspections and if appropriate, other independent reviews and assessments [104]. New inspection methods are also applied depending on their development, new potential for application and if appropriate, on the basis of experience from other reactors.

Surveillance programme

At the time of reactor construction and subsequent reconstructions, the surveillance programme was not considered and implemented for the materials of reactor vessel or internals. For LVR-15 research reactor, a single surveillance programme is in place, focusing on determining the condition and estimating the residual life of the flange joint of horizontal reactor channels. The Programme is part of the document "Life Assessment of Selected LVR-15 Research Reactor Components" [104].

When assessing life of the selected components, corrosion resistance and leak tightness of the flange joint between the vessel shell (its stainless steel flange) and the aluminium flange of the horizontal channel were stated as one of the core issues. Although the inspections confirmed the leak tightness of the horizontal channel, it was deemed necessary to set up a model flange of the horizontal channel, which will be placed directly in the primary coolant flow in the reactor.

The model ("surveillance specimen") flange of the horizontal channel is made on a scale 1:1, from identical materials and with identical technology as actual horizontal channels and is placed in the reactor vessel to simulate as closely as possible the operating conditions, under which the flanges of the horizontal channels work. The surveillance specimen was inserted to the vessel in 2008.

A surveillance specimen will be withdrawn after ten-year exposure in the reactor and then subject of inspection and analysis of the seal, estimating the residual life and if appropriate, defining corrective actions.

Provisions for identifying unexpected effects of degradation mechanisms

The effects of any unexpected degradation mechanisms are identified through the following activities

- Complex testing of the readiness of reactor technology before startup and fulfilment of the relevant criteria for operability
- Inspection of primary chemistry and other tanks as possible indication of increased corrosion in the given environment
- Monitoring of radiation situation in the workplace, monitoring of discharges and environmental monitoring as possible indication of the failure of barriers
- Inspections by reactor operators as part of on-site walkdowns with the potential for detection of abnormal audio/visual conditions and effects

2.3.4 Preventive and remedial actions

2.3.4.1 Preventive and remedial actions – Dukovany and Temelín Nuclear Power Plants

A systematic approach to preventive and remedial actions is addressed under the particular programmes (Plant Programmes, AMPs and Maintenence Programmes identified under the AMR as the Ageing Management Programmes), which are in conformity with the nine attributes of the effective AMPs by the IAEA [6].

In the process of reliability and life management, remedial actions (preventive and corrective) are part of the creation of Health Reports [56] and are assessed according to ČEZ_ME_0889 Evaluation of the State of Assets [68].

2.3.4.2 Preventive and remedial actions – LVR-15 nuclear research reactor

The basic preventive actions are referred in the AMP of reactor LVR-15 as well as in the strategy for reactor operation and internal project, the so-called “re-licensing”, including the identified necessary investment actions associated with the restoration of critical components before the submission of application for license renewal after 2020. The basic preventive actions include gradual equipment modernisation to meet current standards including determination of life and qualification for work environment.

Complementary measures involve strict compliance with the LaC in operation, thus ensuring and demonstrating that equipment is exposed to defined conditions.

2.4 Review and update of the overall AMP

2.4.1 Review and update of the overall AMP for Dukovany and Temelín Nuclear Power Plants

Implementation of licensee’s audit and inspection findings

The review and update of the overall AMP are carried out under the PSR in a ten-year period and under the AMR (at the time of unit preparation for LTO and then every five years). Solving the findings related to safety is covered by the Safety Improvement Programmes for EDU, ETE [57], [58], which are submitted to the State Office for Nuclear Safety once a year.

Evaluation of plant specific experience, external experience, taking account of current state-of-art including R&D results

In addition to the aforesaid methods of use of internal and external experience under the Ageing Management Review, deviations are assessed during standard performance of the programme and recommendations are developed to ensure effective ageing management. Two types of deviations are monitored:

- deviations from the limiting values of monitored parameters,
- deviations from expected state identified in the ageing assessment on the basis of the AMP, in the assessment of the conditions of validity of the TLAA and in the continuous performance and condition monitoring of SSCs.

These deviations are assessed in terms of equipment ageing and a recommendation is developed to ensure effective ageing management. The effectiveness of the remedial actions taken is assessed in the Health Reports. Both the deviations and recommendations from AMPs and TLAAs, and the deviations identified during continuous performance and condition monitoring are assessed, for which assessment in terms of equipment ageing is required.

To evaluate external experience, information obtained under ČEZ_ME_0723 Internal and External Feedback from Operating Experience in NPP [62] is used, which states that external feedback takes experience and lessons learned to ensure its programme from the following external sources:

<u>WANO</u> - event reports on the WANO network

- JIT - summary of information on operating experience that may be regarded as useful for the purposes of PJB;

- Preliminary WER - initial notification of the event within 30 days - it is issued only for the events where it is worth issuing the immediate warning or notice to investigate and to take actions in the plants of other Member States of the WANO; the report should be updated within 140 days of the initial notification;
- WER - detailed report with an analysis of the causes and consequences of the event including any remedial action taken; the report should be published within 140 days of the event;
- SER - Significant Event Report;
- SOER - topical analyses of event reports including defined recommendations;
- Hot Topics – analyses of current areas of operation;
- CEO Updates (hot topics for top management);
- ENR – accelerated reporting to the WANO network

IAEA - event reports on the IAEA network - IRS system
Events from Slovak NPPs – analyses of operational events in the EBO/EMO/JAVYZ
Events from secondary locality Dukovany/Temelín NPPs - analyses of operational events in the Dukovany/Temelín NPP obtained from secondary-locality meeting minutes of the Failure Commission

Publicly available sources of events in non-nuclear industry, e.g. Deepwater Horizon drilling rig event, events in conventional plants, in civil air traffic.

In addition, annual specialised seminars are organised in order to share experience in the area of ageing management and long-term operation of NPPs with the participation of specialists from ČEZ,a. s., SE and ÚJV Řež, a.s., and other specialised companies and institutions, based on the current issues discussed. Furthermore, annual meetings are organised in the area of reliability management - the Czech Republic, Slovak Republic, Hungary and benchmarking (VVER chemistry database - the Czech Republic, Slovak Republic, Hungary and Finland).

Participation in the international activities in this area within the IAEA also serves to monitor and implement the current state-of-art in the area of ageing management. These include specifically:

IAEA SALTO (Safety aspects of long term operation):

a) Employees of ČEZ, a. s., as well as the subsidiary company ÚJV Řež, a.s., participated actively in that extra budgetary programme between 2003 and 2006, in which fundamental principles have been defined for preparation of safe long term operation, focusing in particular on the proper setting of ageing management process. The IAEA Safety Report No. 57 "Safety aspects of long term operation of Nuclear power plants" [7] is the outcome of this project.
b) A total of four IAEA SALTO Peer Review Missions took place in the Dukovany NPP between 2008 and 2016. In the framework of theses missions, plant preparedness for the LTO was reviewed in the following areas:
    - Organization and functions, current licensing basis, configuration/modification management
    - Scoping and screening and plant programmes relevant to LTO
    - Ageing management review, review of ageing management programmes (AMPs) and revalidation of time limited ageing analyses (TLAA) for mechanical components

- Ageing management review, review of AMPs and revalidation of TLAAs for electrical and I&C components
- Ageing management review, review of AMPs and revalidation of TLAAs for civil structures.
- Human Resources, Competence and Knowledge Management for LTO.

c) A total of 3 IAEA SALTO Workshops took place in the Czech Republic between 2014 and 2016, where the experiences of participants from different countries with LTO preparation were discussed.
d) Since 2006, workers of ČEZ, a. s., as well as the subsidiary company ÚJV Řež, a.s., have regularly taken part in the foreign IAEA SALTO Peer Review Missions and the IAEA SALTO Workshops as observers and as experts invited by the IAEA.

<u>IAEA program International Generic Ageing Lessons Learned (IGALL) for Nuclear Power Plants</u>

Since 2009, specialists of ČEZ, a. s., as well as the subsidiary company ÚJV Řež, a.s., have taken part in the IGALL Programme, which is focused on acquiring, maintaining and presenting best practices in the area of ageing management. They are members of both the management bodies of that Programme and all relevant working groups.

<u>EPRI</u>

In 2010, the ČEZ, a. s. became one of the members of the EPRI (Electric Power Research Institute) organisation and acquired the right to use the collected information for more than 40 years of shared research in the area of nuclear energy. Besides the use of available EPRI information and products, the ČEZ, a. s., as one of the organisation members, influences currently the orientation of applied research of EPRI according to its needs.

The membership in EPRI was, in cooperation with ÚJV Řež, a.s., used for implementing more than 20 projects with direct application of knowledge from EPRI, which had technical, safety or economic benefit to improve NPP operation. In many other cases, information from EPRI database was used as supplementary source of information, when it could not be directly applied, e.g. due to different equipment configuration, used materials, valid documentation, etc.

At the same time extensive projects (e.g. Maintenance Optimisation) based on EPRI information and experience were implemented in the company ČEZ, a. s.

ČEZ, a. s., and ÚJV Řež, a.s., are active members of the EPRI and took part in the development of several EPRI documents. In the area of ageing management, they were the main author of the document "Material management matrix for VVER reactors", which summarizes knowledge of degradation mechanisms and areas for further potential research for major equipment of VVER 440 and VVER 1000 power plants.

In the area of ageing management, Czech nuclear power plants won the EPRI technology transfer award for the AMP for Cables.

<u>R&D</u>

At the corporate / group level, the <u>R&D</u> is addressed through three elements:

- Working group for R&D (led by the Research and Development Project Manager) – it has 12 topical areas (across the power industry – from energy production, through alternative fuels up to smart cities and energy savings. The first area is nuclear power industry (under the responsibility of the Head of NPP Technology and Development

department), which is divided into four sub-areas: safety (under the responsibility of the Head of Accident Management department), reliable and economic operation (Head of NPP Technology and Development department), nuclear fuel and radioactive waste management (Head of Nuclear Fuel Purchase department), advanced nuclear systems (Research and Development Project Manager). The working group is a source of ideas for the R&D projects proposed by the group.
- Committee for R&D - approves the proposals for R&D projects.
- Budget for R&D projects.

The guideline "SKČ_SM_0038 Opportunity and Project Management in the Research and Development (R&D) Portfolio" [63] deals with the process – elementary proposals, screening of topics, documentation for approval (R&D Project Plans), etc.

For specific equipment category (A), the parameter of <u>conceptual ageing</u> is evaluated in conformity with [27] on an annual basis. Conceptual ageing occurs due to changes in the requirements for safety, changes in the requirements for equipment and changes in the international standards in the light of new results in science and technology.

Source of information to evaluate that parameter:
- Feedback from the processes of Nuclear Safety and Assets Management Strategy;
- Information of the Licensing department;
- Engineering department know-how.

Evaluation of plant modifications that might influence the overall AMP

Assessment of equipment configuration change is carried out according to the ČEZ_ME_0766 "Assessment of Equipment Configuration Change in NPPs" [64]. NPP Long-term Operation Preparation department (Ageing Management and LTO), which is the guarantor for ageing management, is included as an irreplaceable party commenting on all modifications of equipment falling within the scope of ageing management.

The current configuration of SSCs is one of the inputs to determine the category of SSC and to define the strategy for care of SSCs. This is described in the SKČ_PP_0133 "Assets Management Strategy" [24].

Evaluation and measurement of the effectiveness of ageing management

Evaluation of the effectiveness is carried out at the level of evaluation of the effectiveness of reliability management, by monitoring performance and condition of technological systems and equipment of NPPs under [56] and is documented in the Health Reports for individual technological systems.

Evaluation of the effectiveness is also carried out at the level of individual AMPs (for individual AMPs see Chapter 3 to 8 of this Report).

For more information see also Chapter 2.3.1.1, part "Quality assurance of the overall AMP".

Evaluation of time limited ageing analyses

The evaluation of time limited ageing analyses is carried out according to the procedure [23] and the methodology [29].

Developing TLAAs is one of the possibilities, depending on physical ageing, how to determine the current and predicted state of equipment, its current lifetime and predicted lifetime at the end of planned period of operation.

The developed TLAAs are, according to [23], identified, updated, revalidated, registered and the conditions of their validity are evaluated.

The TLAAs based on the above mentioned methodology meet the criteria (attributes) required in Regulatory Safety Guide No. JB-2.1 [5], specifically:

- They are related to the equipment intended for assessment for LTO
- They take account of degradation mechanisms / ageing effects
- They include time limited assumptions defined by planned period of operation (e.g. according to the valid design - number of years)
- They include conclusions or provide a basis for conclusions concerning equipment capability to perform the intended function
- They can be included as references in the current licensing basis (FSAR)

Activities based on this methodology are summarised into the following four steps:

Step 1 - before proceeding to the development of TLAA, an ageing management specialist decides on the method for monitoring, assessment and ageing management judgement on the basis of understanding of ageing (i.e. analyses containing information on degradation mechanisms, ageing effects, stressors, affecting performance of required equipment function, which should be ensured by ageing management) according to [23]. In this step ageing management specialist should take account of the degradation mechanisms and equipment, for which development of the TLAA should be considered (mainly equipment in the scope of the overall AMP). If this analysis determines that the TLAA will be developed for particular equipment and degradation mechanism, Step 2 takes place.

Step 2 - the TLAA is developed for equipment and degradation mechanisms identified in Step 1. The TLAA should be developed while meeting the requirements for TLAAs (see above).

Step 3 - conditions of validity are verified for the TLAA. It means verifying whether all assumptions, used in the TLAA, are still valid.

Step 4 - the TLAA is properly documented and registered.

Implementation of the current state-of-art including R&D results

The current state-of-art is, according to [23], included through a wide use of the results of research and development, and by using external and internal operating experience (see Chapter 2.3.2.1, parts “Processes and procedures for the identification of degradation mechanisms and their possible consequences”, “Use of R&D programmes” and “Use of internal and external experience”, and in Chapter 2.4.1, part “Evaluation of plant specific experience, external experience, taking account of current state-of-art including R&D results”)

Update of the overall AMP in the light of changes in legislative and regulatory framework

Legislative changes, changes in national guides and international requirements are monitored already at the stage of their preparation (the Atomic Act, Implementing Decrees, SÚJB Regulatory Guides, WENRA SRLs). NPP operator is included in the process of commenting on proposed changes. Possible effects on licensees´ processes are considered already at this stage. New requirements are gradually implemented in all licensees´ processes - e.g. the transition period to meet the requirements of the new Atomic Act is two years of the date of the entry into force.

The general procedure is provided in:
ČEZ_PP_0327 Communication with Authorities [65]
ČEZ_PP_0328 Cooperation in Developing Legislation [66]
ČEZ_PP_0326 Legislation Application [67]

Identification of needs for further R&D

For detailed information see Chapter 2.4.1, part "Evaluation of plant specific experience, external experience, taking account of current state-of-art including R&D results".

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

The review and update of the overall AMP are carried out within the framework of the PSR with a ten-year period and within the framework of the AMR.

The system of the overall AMP is reviewed under the PSR; the current state of ageing management is reviewed under the AMR including assessment of physical ageing for defined groups of SSCs.

Incorporation of unexpected or new issues into the AMP

For the entire scope of SSCs in the AMP, incorporating unexpected or new issues is carried out in the framework of continuous performance and condition monitoring of SSCs according to [24], which includes inputs from internal and external feedback. In paragraph "Identifying Solution to the Unsatisfactory Condition" – possible methods of solution to the unsatisfactory condition are defined, e.g.:

- *Decision on change of the SSCs care strategy - a change in the scope of the preventive maintenance programme (update required, or new maintenance template), update of the AMP/ TLAA, new AMP/ TLAA, changes in specific SSCs of category A;*
- *Requirement for a change in the categorisation - reassessment of the categorisation based on the discrepancy identified in the functional role of SSC and the current categorisation of SSC;*
- *Requirement for a change in monitoring - requirement for a change in monitoring parameters, or a change in limit values of monitored parameters aimed at better control of unsatisfactory condition or trend of SSC.*

For specific equipment category (A), incorporating unexpected or new issues into the AMP is part of the evaluation of the parameter "Conceptual Ageing" according to [27]:

- *Information from feedback and information from Licensing department (information on changes in regulatory requirements) is assessed against the real condition of plant and associated documentation binding to operate equipment with the following assessment alternatives:*
  - *The rate of conceptual ageing is acceptable with no additional recommendations;*
  - *The rate of conceptual ageing requires an analysis of the condition and the proposal for solution to ensure the required level of parameter (issuing the technical suggestion, revision of the Component Specific Ageing Management Programme or Specific AMP, another proposal for solution).*
    - *The results are presented in the annual equipment's life assessment and subsequently in the relevant Health Report.*

- *In the framework of comprehensive equipment's life assessment, a life management specialist proposes a solution to ensure the required level of conceptual ageing.*

Review of the overall AMP on the basis of results from monitoring (operating parameters), operation testing, sampling, and inspection activities

The programmes for monitoring of operating parameters, operation testing, sampling and in-service inspection activities are part of the ageing management process and, where necessary, their results are part of the standard feedback of the overall AMP.

Periodic evaluation and measurement of the effectiveness of ageing management

Periodic evaluation and measurement of the effectiveness of ageing management are described in Chapters 2.3.1.1, part "Quality assurance of the overall AMP" and 2.4.1, part "Evaluation and measurement of the effectiveness of ageing management".

2.4.2 Review and update of the overall AMP for LVR-15 nuclear research reactor

The overall AMP will be fully implemented in compliance with new legislation as part of the harmonisation of reactor operating documents and SÚJB Decree No. 21/2017 Coll., by the end of 2018. The Programme will be implemented with the use of the results of reactor inspection plan including the so-called "five-year" inspections carried out in 2017; evaluation of the surveillance specimen of flange joint in 2018; case-by-case analyses of the condition and feasibility studies from individual projects of gradual modernisation, and with the use of agency guides SSG-10 and the Regulatory Safety Guide for research reactors [8] prepared by the State Office for Nuclear Safety.

On the basis of harmonisation with legislation, the process will be supplemented by a management system programme including assignment of responsibilities for the review and update, and the requirements for independent assessment and the system of audits in accordance with the company's management system and quality system. Another independent review of the process and setting of the ageing management system will be part of the invited INSARR mission for 2019.

2.5 Licensee's experience of application of the overall AMP

2.5.1 Licensee's experience of application of the overall AMP for Dukovany and Temelín Nuclear Power Plants

Lessons learned during operation showed the need for changes in both the organisational structure and the scope and structure of the overall AMP. The ageing management in the Dukovany NPP and the Temelín NPP was initially based on manufacturers' recommendations and if appropriate, on the state-of-art in the area of ageing management at that time (Dukovany NPP 1985-1987, Temelín NPP 2000-2002). There were particular actvities of ageing management such as surveillance programme, in-service inspection programme, maintenance programme, and operating mode monitoring programme. However, ageing management was not ensured comprehensively.

On the basis of lessons learned and good practice identified, individual Ageing Management Programmes (e.g. Monitoring Programme for Pipings Affected by Flow-accelerated Corrosion; Ageing Management Programme for Cables Important to Safety; Fatigue Damage Assessment Programme) were gradually implemented.

After 2000, first projects began, aimed at implementing a comprehensive approach to life/ageing management:

- Research project of the Ministry of Industry and Trade and ČEZ – methodologies and basis for the SW environment for the overall AMP were developed in this project.
- First assessment of the current state of life management for Dukovany NPP/Temelín NPP (it served as a basis for developing the Specific Ageing Management Programmes and the Component Specific Ageing Management Programmes for category A equipment).
- Effective maintenance strategy project – ensured maintenance optimisation aimed at management of identified failure modes.
- AMR EDU – ageing management review for LTO.

Changes in the organisation in the area of ageing management took place in parallel with these projects.

The original departments dealing with ageing management were separate for individual sites in the framework of management areas called Technical Safety and Care of Assets. Since 2009, there has been a central department dealing with ageing management under Engineering Department which acted as technical support for Care of Assets department.

The initial scope of equipment within the scope of ageing management was given by listing (safety-relevant difficult to replace equipment). At the start of the stage of preparation for operation in LTO (Dukovany NPP), the ageing management was set for equipment important to safety and related to safety on the basis of the review concerning understanding of ageing for both sites.

After gradual modifications, the current setting of the overall Ageing Management Programme is considered sufficiently comprehensive. In operator´s opinion, the overall AMP meets all the currently valid requirements of national regulator and international recommendations, and is therefore considered adequate.

2.5.2 Licensee's experience of application of the overall AMP for LVR-15 nuclear research reactor

The ageing management system was not explicitly required by law until 2017 but this process was included in quality assurance program. On the basis of the INSARR mission and according to the recommendations defined in agency guides, first ageing management programmes were developed in 2008, followed by revisions. The ageing management system is based on inspections and planned reconstructions, modernisations and major repairs of selected reactor components, which are caused by the need to restore functional characteristics of the equipment in question or in order to increase the comfort of operation, reliability, radiation safety and work quality.

In accordance with the new Atomic Act [2], the Ageing management programme for reactor LVR-15 should be modified according to the SÚJB Decree No. 21/2017 Coll. [4] together with defining the precise criteria, setting out the detection and monitoring systems of degradation mechanisms, establishing acceptance criteria and other requirements. The application of the ageing management system faces a challange related to the identification of component life with regard to the former standards from periods of construction and reconstructions, when a substantial part of systems does not have the design life specified or the system of qualification criteria for work environment and working conditions is not defined in accordance with the current standards.

2.6 Regulatory oversight process

Regulatory activities of the SÚJB include the whole range of specialised activities, of which the most important are inspection and assessment activities. The principle of graded approach is applied in their planning.

2.6.1 Process for regulatory oversight of Dukovany and Temelín Nuclear Power Plants

The approach of the license holders for the Dukovany and Temelín NPPs is assessed on a regular basis under the review of the annually updated FSAR.

This area was further assessed by the SÚJB in the assessment of the conclusions of PSR 30 EDU and PSR 10 ETE, the outcomes of which were, in accordance with the conditions of the Decisions on Operation issued, submitted to the SÚJB.

For the Dukovany NPP, the application of the entire approach was verified in detail under the review of licensing documents demonstrating the potential for continued safe operation of the Dukovany NPP after 30 years ("LTO").

Furthermore, the SÚJB verifies and evaluates information concerning ageing of individual SSCs as part of its inspection activities.

2.6.2 Process for regulatory oversight of LVR-15 nuclear research reactor

The SÚJB evaluates information concerning ageing of LVR-15 research reactor components as part of its periodic inspection and assessment activities through review of the FSAR and other information from operation of that facility and also as part of its inspection activity. In this activity, the Office particularly focuses on the most significant components.

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

2.7.1 Assessment of the overall ageing management programme for Dukovany and Temelín Nuclear Power Plants

The overall Ageing Management Programme of ČEZ, a. s. has been assessed having regard to the requirements of the Atomic Act [2] (see Chapter 2.1) as well as having regard to the international standards and WENRA Safety Reference Levels [3]. The international requirements concerning ageing management were fully implemented in the national legislative and regulatory framework on 1 January 2017, when new Atomic Act [2] entered into force. Transitional provisions are included in the Atomic Act, which provide the holders of license the time limit to adapt to the new legal conditions (in general, two years). The Atomic Act introduces new requirements for ageing management process as well as the Ageing Management Programme, which were, before that period, included in Regulatory Safety Guide No. BN-JB-2.1 [5]. Even though setting of the approach to ageing management matched the international good practice, the entire process is now being adapted to meet the new Atomic Act within the license holder´s managing areas.

The SÚJB verified in detail the ageing management process of ČEZ, a. s. in the period of assessment of supporting documentation for safe operation after 30 years of operation ("LTO") of the Dukovany NPP (before validity of the new Atomic Act). The deficiencies found, which related to the implementation of the overall ageing management programme for individual SCs and the requirements to remedy the deficiencies found were imposed in the framework of the conditions of Decisions on Operation of Dukovany NPP Units (for details see chapter 9). The overall AMP is also subject to assessment of the results of periodic safety review for both power plants. This review is carried out every 10 years; in the Dukovany NPP, the PSR after 30 years of operation was terminated in 2013 and no serious deviation has been identified. In the Temelín NPP, the PSR relating to 20 years of operation will be carried out between 2018 and 2020. In the Temelín NPP, all activities are currently not terminated, which are a precondition for full implementation of the overall AMP (e.g. completion of the AMR for all components (i.e. not only selected equipment) included in the scope of equipment subject to ageing management); the implementation will be terminated by 2018.

2.7.2 Assessment of the overall ageing management programme for LVR-15 nuclear research reactor

An analysis was carried out in the Ageing Management Programme for the LVR-15 nuclear research reactor, in which possible degradation mechanisms affecting selected equipment were analysed and ageing effects were identified together with an assessment of the possibility of equipment replacement. On the basis of this analysis, equipment affecting nuclear safety was identified, which is difficult to replace and the detection methods were also assessed. Remedial actions were defined to mitigate the effects of ageing. A draft measure is the outcome of the Programme, which serves as a basis for additional inspections, maintenance and reconstructions of systems. The Programme does not specify rights and obligations of the workers involved in the ageing management process or feedback provision to assess or increase the effectiveness of the Programme. There are not activities stated, concerning the prediction of the future condition of components.

Given the newly introduced requirements in the current legislation valid from 2017, this Programme will be (with the use of transitional provisions) revised by the end of 2018 to meet all attributes of effective Ageing Management Programme, which are specified in new legislation.

3. Electrical cables

3.1 Description of ageing management programmes for electrical cables

3.1.1 Scope of ageing management for electrical cables

3.1.1.1 Scope of ageing management for cables of the Dukovany and Temelín Nuclear Power Plants

Brief history of ageing management of cables

In Czech nuclear power plants, work on the component specific Ageing Management Programme for Cables (hereinafter referred to as the "AMPC") for cables important to safety [69] began in the second half of the 1990s. The component-specific AMPC [69] is based on the experience with cable qualification for the environment of the Dukovany and Temelín NPPs and on the international recommendations of the IAEA, EPRI, OECD/NEA and other documents (the most important documents are referred in references to this Chapter). The component-specific AMP for cables and all associated activities are also described in the related methodologies - specific AMPs related to cables [70] and [71] - see also Chapter 2, the outcomes of which are integrated into the component-specific AMP for cables (AMPC). All references, documents and standards are also referred in these methodologies [70] and [71], by which the AMPC is established and has been implemented since 2006. The Programme is updated on a continuous basis as required by the operating organisation and international good practice.

At the beginning, the AMPC was oriented on low-voltage (LV) safety cables with the requirement for the functionality during design basis accident (DBA). The AMPC currently includes all important to safety cables. From the beginning of the AMPC in the Czech NPPs, there is intensive international cooperation between the IAEA, EPRI, OECD/NEA and other organisations for the purposes of its continuous improvement.

The main attributes of the AMPC are as follows:

- The Programme began more than 20 years ago
- Cables of the Temelín NPP were included in the AMPC before the commissioning of the plant
- Ageing and condition of cables are assessed by means of several hundreds of surveillance cables located in tens of cable deposits at different locations of both plants in the Czech Republic
- Obtaining detailed knowledge of the parameters of the environment. Temperature, dose rates and moisture have been measured at hundreds of locations in the NPP since 1996
- Cables in operation are subjected to periodic visual inspections
- Cables removed from the NPP during technology restoration are subject to assessment
- Functional feedback is established between NPP equipment managers and experts implementing the AMPC
- A database system is in place for the archiving of data related to cables, environmental parameters and outputs of visual inspections and calculations of the residual cables lifetime

- Periodic annual evaluation is carried out for safety cables operability and the residual lifetime of cables is being specified with the outputs for the Health Reports, Periodic life assessment and FSAR

Selecting cables for the AMPC

The AMPCs in Czech NPPs include all cables important to safety. These are the cables connecting the safety systems and the safety-related systems. The selection is also in accordance with the IAEA terminology and with the classification according to ČSN EN 61226 [72].

The comprehensive selection of cables includes all types of safety cables, i.e. communication, power, low-voltage, coaxial, high-voltage, optical, etc., cables as well as cables of different constructions. The SSK (Cable management system) database system serves as a primary source for selecting cables important to safety for the AMPC, which is operated in both Czech NPPs. The SSK is a software application for designing and management of the cable systems of large-size technological units containing data on the actual condition of cable rooms and the cables installed therein. The SSK always contains data related to all cables. The SSK, depending on the connected equipment, provides information whether or not the cable is one of the cables important to safety. The aggregation of the number of cables in the AMPC is indicated in Table 3.1; the list of types is provided in Tables B.1 and B.2 in Annex B. The list of safety cables is, for the Ageing Management Programme, updated in the light of data obtained from the SSK at least once a year.

The safety cables are divided into three main categories by requirement for their functionality during normal operation and accidents:

1. Required functionality during design basis accident (LOCA/HELB) - the cable has to be qualified according to the appropriate standards [73], [74], [75], [76]
2. They can be exposed to LOCA/HELB (harsh environment); the functionality is required only untill the beginning of the incident - qualification for accident conditions is not required.
3. Cables of the safety systems that will not be exposed to the conditions of DBA (mild environment) - qualification for accident conditions is not required.

Table 3.1: Aggregated data on the number of safety cables and types included in the AMPC in the Czech Republic

NPP	Number of units	Number of safety cables	Types of safety cables
Dukovany	4	63,717	113
Temelín	2	31,570	82

Brief description of the AMPC

The main objectives of the AMPC are as follows:

- Assessment of the condition, operational ability of the safety cables by types in all rooms, where the safety cables are installed, in both NPPs

- Determination of the residual lifetime of safety cables in the NPP together with recommendation onhow long can the cable be safely operated.

Basic activities under the AMPC include:

a) Summary of all information as regards cables in operation:
   - Cable manufacturers, specifications, constructions, core and jacket insulation materials, types of ageing tests carried out for cables
   - Basic characteristics of the cables
   - Changes in the characteristics of cables over time; i.e. accelerated ageing testing, measurement of mechanical, electrical and physico-chemical parameters, determination of activation energy
   - Cable routing in NPP rooms
   - Parameters of the environment in which the cables are laid
   - Input information for the Qualification Programme
b) Measurement of the environmental parameters
   - Temperatures
   - Dose rates
   - Moisture
   - Neutron fluxes
c) Condition assessment of the cables in operation:
   - Measurement of mechanical and electrical parameters of the cables taken out of service
   - Visual inspections of cables in the NPP
d) Measurement of the mechanical and electrical parameters of cable deposits. There are more than 400 surveillance cable deposits installed in the NPP, with different environmental parameters, located in both harsh environment (temperature > 40°C, dose rate) and mild environment (temperature below 40°C, no dose rate)
e) Calculation of the residual life of cables on the basis of the summary of lessons learned from the activities carried out under the AMPC.
f) Preparation of the surveillance specimens for the AMPC, for NPP operation beyond the design limit and for any qualification tests combined with the simulation of DBA
g) Care of cable deposits

General solution to the AMPC is shown in the diagram in Fig. 3.1 (page 56). For better orientation, cables are divided into three main categories in the diagram.

1. Cable in operation
2. Cable deposit
3. Calibrated cable

<u>Cable in operation</u>

All activities under the AMPC are directed towards the safety cables in order to determine their current condition and to predict their life durability.

To determine the residual lifetime, the following is required:

- List of cables, installation date.
- Complete routing of all cables.
- Temperature and dose rate in every room where the cables are laid.
- Knowledge of the effects of environmental parameters on the rate of ageing.
- Algorithms and models used in the determination of life durability.

Condition assessment of the cables in operation is supplemented by:

- Periodic visual inspections of cables.
- Use of the results of the condition measurement of cable deposits.
- Measurement of the samples taken from operation during the replacing of cables.

Cable deposit

Cable deposits (surveillance cables) are cables that age at different locations of the NPP. They are regularly assessed, and changes in the characteristics may predict the condition of cables operated under the same or similar conditions; by conversion under different conditions. In addition to the environmental conditions that occur near the main circulation piping (i.e. the worst environment that occurs in the place of cable operation in the NPP), cable deposits are placed in approximately 30 other cable deposits with different environmental parameters. More than 400 in total surveillance cables ages in both NPPs. A typical sample of the surveillance cable (deposit) is as follows:

- Several meters for electrical tests and if appropriate, other non-destructive tests that are carried out directly in the nuclear power plant.
- Several short samples for continuous measurement of mechanical properties and other destructive tests in the laboratory
- Where enough amount of cable is available, there are also several meters for future use, e.g. for assessment in the long-term operation (LTO), re-qualification, etc.

Calibrated cable

Calibrated cables are safety cables that are used in the laboratory during analysis of materials and during accelerated tests, which show changes in functional characteristics depending on temperature, dose, dose rate and ageing time.

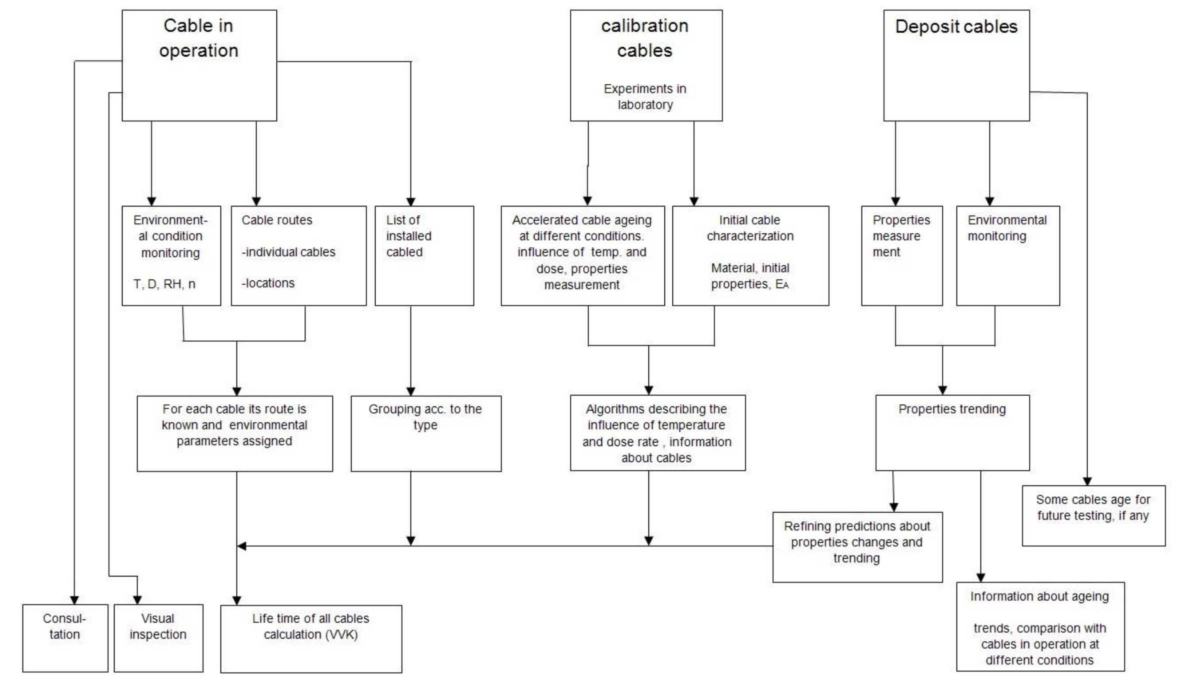

Fig. 3.1: Basic scheme of cable AMPC

According to Chapter 03.1.1 [1], ageing management should be assessed for the following groups of safety cables in the National Assessment Report:

1. High-voltage (HV) cables subjected to adverse conditions (conditions adverse for the lifetime of cable components such as moisture, radiation or temperature). For the purpose of this National Assessment Report, high-voltage cables are those above about 3 kV
2. Medium-voltage (380 V to 3 kV) cables (in trenches or in cable ducts).
3. Neutron flux instrumentation cables.

Given that the AMPCs in the Czech NPPs include all safety-related cables (Table 3.1), the group covers automatically the cables required in points 1 to 3 provided that they are classified as safety cables.

On the basis of different voltage levels or other special requirements for the segregation, the cables in the Czech NPPs are included in several segregation groups. Each cable in the NPP has the segregation group indicated in its unique identification code and selected groups can, therefore, be filtered out of the SSK system. The following groups are the current groups for the cables required in Chapter 03.1.1 [1]:

Table 3.2:

Segregation group	Description
WA	Power supply cables at high voltage level
WB	Power supply cables at low voltage level (new cables, transmitted power below 5 kW may be identified WL)
WL	Power supply cables at the level of 380/220 (or 400/230V) VAC or 220VDC (transmitted power ≤ 5 kW)
WX	DNIS system cables (neutron flux instrumentation; voltage level 600 V, 850 V)

<u>High-voltage cables</u>

Brief summary of HV cables selected according to the SSK system (the WA segregation group is referred to in Table 3.3.

Table 3.3: High-voltage cables in the Dukovany NPP and the Temelín NPP

Number of HV cables	Dukovany NPP	Temelín NPP
Total number of HV cables	801	957
Overall length of all HV cables	151 km	154 km
Number of types	16	17
Number of safety cables HV	443	455
Overall length of safety cables HV	63.5 km	56 km
Number of HV types of safety cables	10	2

The overwhelming majority of high-voltage cables in the Dukovany NPP are operated under the mild conditions with temperature below 40°C and with no radiation. Only two types of safety cables, of the total number of 119 pieces, are placed under the harsh conditions. These are the PVSG (the former Soviet Union) and 6-AYKCY cables (Kabelovna Kladno). The worst operating condition are in the Steam Generators box where there are locations with temperatures of 60°C and dose rates of 0.1 Gy/h. For these cables, functionality in DBA is not required.

There are six types of high-voltage safety cables in the Temelín NPP. One type KUHSC (Alcatel Cable) is used under the harsh conditions in containment. It is routed from the MCP room (GA504/1,2,3,4) to the room of penetrations (GA315/1,2,3). The ambient temperature in these rooms is below 35°C with negligible radiation. For these cables, functionality in DBA is not required.

<u>Medium-voltage cables (380 V to 3 kV), in trenches</u>

Czech legislation does not use the term “medium voltage cables”, as stated in Chapter 03.1.1 [1]. Cables below 1 kV are classified as low-voltage cables, when the Czech NPPs use voltage below 400 V and then high-voltage. The values between the two values are used in extremely rare cases, e.g. for neutron flux instrumentation cables (see chapter below). Cables from the WB or WL segregation group fall within the required category.

In the Dukovany NPP or in the Temelín NPP, there are no power supply cables in trenches with the voltage level of 380 V to 3 kV that are classified as safety cables.

Neutron flux instrumentation cables

In the Dukovany NPP, there are 247 cables used for neutron flux instrumentation system, of which 216 cables are the safety cables. There are five types used: Habia Cable 1-410527 B (this is the only type that is not safety type), Pirelli CP711, Pirelli CP597, VCXJE-V (Kabelovna Kabex), TKC (Mirion, USA). The worst operating conditions are in the Steam Generators box where there are locations with temperatures of up to 60°C and dose rates of 0.1 Gy/h. However, there is only one mineral insulated cable TKC under those conditions. All the other cables are operated undermilder conditions.

There have been 151 neutron flux instrumentation cables identified in the Temelín NPP. There are three types used: KJB (Alcatel Cable), 3A98892H02 (BICC Brand-Rex Company), 4A07470H01 (Chromatic Technologies).

Other safety cables

Cables mentioned in Chapter 03.1.1 [1], serving as a model for a detailed description of ageing management for electrical cables, are only a small part of all safety cables that are monitored and assessed under the AMPC in the Czech NPPs. In the Czech Republic, cables under the AMPC are not assessed by categories; ageing management takes place generally for all safety cables. Therefore, there is no division into categories in the following text describing AMPC information about all safety cables.

Ageing management of the components of cable systems

Conductors: Metallic materials of conductors do not age from environmental effects. Under the IGALL project [22], which is coordinated by the IAEA of which the ČEZ company is a member, it was stated that ageing management under the AMPC is not necessary for conductors. As regards the maintenance of the functionality of conductors, corrosion effect due to increased moisture or due to action of chemicals (e.g. hydrochloric acid from PVC cables) should be taken into account. Corrosion can affect the functionality of communication or coaxial cables. This mainly relates to cable terminations and may be well identified during regular maintenance, diagnostic measurements, visual walkdowns, and may be prevented by sufficient protection (cover) of equipment. In most length of the cable, conductors are well protected against corrosion by insulation provided that it is in good condition. The conditions for maintaining the function of conductors, abilities to conduct an electric current, are therefore generally fulfilled or may be fulfilled during regular maintenance on equipment.

Shield: Shield is made from metallic materials. Similar to conductors, there are no special ageing management programmes for shielding [22]. Corrosion can affect its function in maintaining sufficient EMC shielding or can affect field distribution for high-voltage cables.

Armouring: It provides further protection for cables. Operating conditions do not virtually affect the ageing.

Insulation of cable conductors and jackets: In the vast majority of cases, they are made from polymer materials, where a number of external or operational influences activates or accelerates their degradation. These are sensitive parts that are under primary attention during the AMP.

Termination arrangements: These parts include connectors, cable terminals and cable joints. The termination type can be found in the SSK. There is small number of safety connectors in nuclear power plants typically. This is due to a general requirement of the NPP that termination arrangements for safety systems should be made with fixed termination, where possible.

Connectors are used only to the extent necessary, mainly in the I&C system cabinets. These are the connectors, for which the functionality in DBA is not required; they are mostly located outside the harsh conditions. The only exception are qualified connectors of the SNC, LEMO and ILF 14c types in the Dukovany NPP and of the SNC, LEMO, Veam and Westinghouse types in the Temelín NPP . It is always a part of the whole system, e.g. temperature-monitoring system in the Temelín NPP with SNC and Westinghouse connectors. Connectors are not assessed under the AMPC but under preventive and corrective maintenance on particular equipment or systems. Other activities, except for electrical functionality testing (e.g. contact resistance, insulation resistance), involve the regular exchange of rubber seals, the condition of which is one of the main factors affecting life durability. In addition, the SNC and Westinghouse type connectors were intensively studied under the AMPC and during qualification tests carried out in the company ÚJV Řež, a.s.

The present report covers the cables and therefore, cable arrangements are not described herein.

Identification of ageing mechanisms related to cables

Polymers are the most used insulation material for cable systems (polymer insulation of cable cores and jackets).

Identification and specification of the ageing mechanisms are based on the following:

- Knowledge gained under the long-term solution to the AMPC as well as qualifications of cables, connector couplings for nuclear power plants in Europe and Asia
- Participation in international groups or IAEA projects (e.g. IGALL), OECD/NEA and EU
- Relevant study research

Degradation mechanisms are described, including their relevance and management option, in many international documents (see references) and in ČEZ documents referred to in Chapter 2. For more detailed information see Chapter 3.1.2.

3.1.1.2 Scope of ageing management for cables of the LVR-15 nuclear research reactor

Cables of the systems most relevant to safety of the LVR-15 research reactor are as follows:

1. I&C that is divided into:
    a) PCS (protection and control system)
    b) I&C (instrumentation and control)
2. EPS1, EPS2 (Emergency power supply)

I&C cables

PCS

In 2016, a new PCS was put into operation, which completely replaced the old system including cables. New cables meet the qualification requirements of current standards and meet the following criteria:

- Be properly classified;
- Be properly sized;
- Have the required fire-technical features;
- Comply with the conditions of the environment, in which they will be laid;
- Have the construction corresponding to its intended purpose (requirements for the type of transmitted signal, accuracy, EMC immunity, mechanical resistance and strength, etc.);
- Routing of these cables should respect the requirements for separation and segregation.

I&C

The current I&C system was put into operation in 1989.

Separate measuring circuits, fully independent of in-process measurements, are installed in the I&C system to control the exceeding of limit values of the parameters of cooling circuit 1, specified in the LaC. Also, electrical power supply for supply power to instruments of these circuits is separated and comes from the systems of essential power supply EPS 1, 2.

These circuits are marked as selected circuits and are classified as Safety Class 2. Information about the exceeding of maximum limit value or the drop below minimum limit value is transmitted as warning to the failure warning system of the PCS and as emergency signal to the circuits of the reactor protection system in the PCS resulting in reactor shutdown.

To interconnect the individual elements of selected measuring circuits, the following cable types are used:

- Signal cables – NCEY, TCEKE
- Power supply cables – CYKY

Cables from sensors on primary circuit pipework are routed in a separate sheet metal cable tray outside the area of the primary circuit to separate frames. Cables are routed from these frames from the ground level to the operator room on the second floor. Cable routes are horizontally and loosely routed mainly under the floor of rooms and corridors in special concrete cable trays.

Cable routes are vertically routed from the ground level to the second floor through the space of separate recesses in corridor walls. In these areas, they are mounted on special cable racks. Cable routes are physically separated from other media and systems.

Emergency power supply system (EPS 1,2)

The system of redundant power supply for the LVR-15 research reactor, to ensure power supply for safety systems, was substantially reconstructed in 2003 and 2007 in accordance with applicable standards and safety requirements.

The concept of EPS modifications was based on experience in the field of EPS for power reactors.

The reconstruction included replacement of power cables.

Operating conditions

Cables of all the systems concerned are located in the common environment, according to the protocol to determine the environment. Thus, they are not exposed to any extraordinary influences that would significantly accelerate degradation of the properties of circuits.

Environmental parameters:

Maximum range of temperature cycles:	15-35°C
Maximum pressure:	100 kPa
Maximum moisture:	50%

Cables are not part of the Ageing Management Programme for the LVR-15 research reactor; their condition is monitored under different programmes of the plant, e.g. In-service Inspection Programme. Cables are readily accessible for inspection (most cables on a daily basis). In addition, the functionality of cables is verified as part of functional checks on equipment, to which those cables belong. The lifetime of PCS, I&C and EPS 1,2 cables is determined in the qualification of cables for the conditions in which they work, until 2030.

3.1.2 Ageing assessment of electrical cables

3.1.2.1 Ageing assessment of electrical cables of the Dukovany and Temelín Nuclear Power Plants

In this Chapter, the AMPC is specified, with an overview presented in Fig. 3.1, in terms of the description of degradation mechanisms, their monitoring, cable assessment and use of operating experience from Czech NPPs and world practice.

For the purposes of ageing assessment and determination of the residual lifetime of cables, it is necessary to know the influence of degradation mechanisms on the ageing process, to find out what degradation mechanisms affect the individual cables at different locations of the NPP and to establish a suitable criterion to estimate the residual lifetime. Model representation of cable lifetime is shown in Fig. 3.2. Critical condition means a cable condition in achieving the acceptance criterion, i.e. value of the selected property (including safety margin), beyond which the rate of degradation is so significant that it can jeopardize the capability of cable to perform its functions in normal and mainly emergency operation.

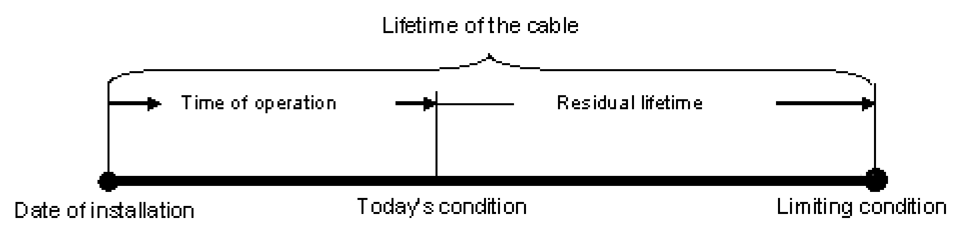

Fig. 3.2: Time representation of the residual lifetime.

Identified degradation mechanisms and ageing effects including identification of their significance

Polymers are the most used insulation material for cable systems (polymer insulation of cable conductors and jackets). For these materials, regardless of function, the long-term ageing process takes place, during which there are changes in the properties, which generally lead to deterioration in the functional characteristics of cables and their insulating and mechanical properties.

The list of degradation mechanisms, their description, effects on the functional of cable / entire system and their control option are described in detail, e.g. in documents [77], [22], [78]. Furthermore, an extensive table was produced, which summarizes degradation effects for individual I&C component (cables, connectors, penetrations, drives, sensors, etc.), significance of individual degradation effects, kinetics, their control option and other information [22], [79].

Degradation mechanisms are presented below in the context of ageing effects on cables:

- Degradation due to ionising radiation
- Temperature effect
- Moisture, steam, water
- Catalytic reaction
- Action of chemicals
- Joule (ohmic) heating
- Mechanical stress

Temperature effect and ionising radiation effect are the most significant degradation mechanisms affecting cables. They are intensively monitored as part of the determination of environmental parameters, see below. The effect of these monitored quantities on the degradation process is analysed for all safety cables in the laboratory or in the deposit.

Other effects, although they are also tested (e.g. mechanical stress), are not included in the most significant degradation mechanisms. These degradation mechanisms are monitored during periodic visual inspections of cable systems. This is in accordance with international good practice [78], [79], [80], [81], [82], [87], as well as with the recommendations under the IGALL [22].

Degradation of cable insulations is most manifested by their embrittlement. Maintenance of the elastic properties of insulations is directly linked to their capability to retain mechanical integrity and geometric dimensions, even in case of movements during maintenance, minor modifications in the cable distribution systems or transfer of stress during installation (bends, clips, etc.). Material embrittlement may be most easily assessed by mechanical tests, measurement of elongation at break. New methods for cable condition assessment and determination of residual lifetime are developed and tested in the world and in the Czech Republic, which are based on measurement of electrical properties over the total length of cables. They are usually based on measurement of electrical parameters and their comparison with laboratory-aged samples. The aforementioned analysis shows that elongation at break of conductors and jacket insulations is the essential criterion for degradation monitoring of cables.

Acceptance criteria

Criteria (including safety margin) are set, beyond which the rate of degradation is so significant that it can jeopardize the capability of cable to perform its functions in normal and mainly accident operation. New cable should meet the manufacturer's specifications or NPP requirements. Some properties can change during ageing. The cable must not show any damage to the extent that could compromise its functionality. The acceptance criteria should be conservative in order to cover potential material non-homogeneities, uncertainties in the assessment methods applied, etc.

The recommended acceptance criteria are referred to in the IAEA documents [78], [80], EPRI documents [82], [83] and other international references [81], [84], [85], [86], [87] for the purposes of applying them in the AMPC.

In the Czech NPPs AMPC, the following criteria are applied:

1. Elongation at break of jacket and conductors insulations is the essential acceptance criterion for cables. Its value must not fall below 50% for safety-related cables.
2. Defining simple absolute criteria for electrical parameters only makes sense for specific applications and cable types (constructions). General acceptance criterion for all cables should be defined with a large margin, to cover all possible alternatives. However, this would disqualify the vast majority of cables.
3. Methods for combining measurements of several electrical parameters offer the relevant potential to determine cable condition with the possibility of indicating changes (degradation). It is necessary, however, to carry out long-term measurements and set the parameters individually for specific cables. The advantage is that it can be applied to inaccessible cables.

For high-voltage cables, testing partial discharges and dissipation factor tg δ is the appropriate method to determine cable condition.Note: There is currently not known any other chemical, physical or electrical method that would make it possible to clearly define an absolute acceptance criterion for a wide range of cables.

Application of lessons learned from domestic and world practice

Manuals, guides, recommendations of the world organisations EPRI, NRC, IAEA, OECD/NEA and the relevant standards, see References, are used for assessing an approach to monitoring of the condition of cables and their functions under operating conditions of the NPP.

Globally recognised experience with qualification of cables and their components for domestic and foreign NPPs, and solution to the AMPC for domestic NPPs are used in the implementation of the AMPC. The AMPC in the Czech Republic also respects the results of research and development in the field of NPP cable sets and the requirements for ageing management of cables at home and abroad. ČEZ Group’s workers implementing the AMPC regularly attend conferences connected with that topic, co-organised two international conferences on NPP cables [85], [86].

The following are some of the projects in which specialists implementing the AMPC actively participated and the outputs of which were used for developing and updating the Programme:

- FP6 Euratom Project “Management of I&C component ageing in nuclear power plants” – MAGIC, 2007-2008
- FP 7 Euratom Project “Ageing Diagnostics and Prognostics of low-voltage I&C cables” ADVANCE, 2011-2013
- IAEA Research co-ordination program on Management of ageing of in-containment I&C cables, 1998-2000

- IAEA research coordination program on Qualification, Condition Monitoring and Management of Ageing of Low Voltage Cables in Nuclear Power Plant Life Management, 2012-2015
- Influence of pigments on cable life time, IAEA research project, ÚJV Řež, ČEZ and University of Tokyo, 2011-2016
- Project “Enhancing the Capabilities of National Nuclear Institutions to Ensure Safe Nuclear Power Programmes”, INSC Project CH3.01/10, based on cooperation between the European Commission (EC), DG Development and Cooperation – EuropeAid, on one side, and the People’s Republic of China on the other side. Task leader on Equipment qualification, 2014-2017
- OECD/NEA Cable ageing project (SCAP), 2007-2010
- IAEA Expert Missions on Cable Ageing Management Programs for Atucha (Argentina), Angra (Brazil), Qinshan (China), Metsamor (Armenia), Laguna Verde (Mexico) NPPs, 2011-2017
- TA02010218 Project of the Technology Agency of the Czech Republic; Research of cable polymeric material degradation and development of methods for verification of material qualification in the conditions of severe accident of new generation nuclear power plants, 2012-2015
- MPO 7 Legislation FT TA4/0069 - Safety and legislative aspects of the construction and commissioning of new generation NPPs for the power industry in the Czech Republic, stage 10: Fire effects on the qualification of cable systems, 2007 - 2010
- Effects of mechanical stress on the life of NPP cables including resistance in DBA; project financed by the ČEZ company, 2007-2008

Use of internal and external operating experience

International operating experience has been implemented in the documents issued within the international organisations such as IAEA or OECD/NEA [78], [80], [81], [82], [83], [84], [85], [86]. These documents were developed in a wide discussion forum of NPP personnel from the whole world and are regarded as a sufficiently representative basis for the AMPC for NPPs in the Czech Republic. On the basis of questionnaire-based investigation and personal consultations, further more detailed information is obtained on the ageing management programmes, operating experience with cables and on the procedures for the replacement of safety cables in the nuclear power plants operated in the world, including reasons for and scopes of such replacements. The NPPs involved were, for example, plants in Canada, Japan, Switzerland as well as VVER type NPPs in Armenia, Ukraine or Russia. Different cable types and materials are installed in old NPPs in the world (except VVER). Therefore, the results of their ageing can be used only as information to improve general knowledge. However, samples of old cables from Ukraine were also obtained and used for comparing the quality between the two countries.

Further information on cables and their ageing was obtained from operators of Czech conventional power plants, including a sampling of old cables in operation for the purposes of assessing their condition. These findings were used mainly for the safety cables in the mild environment.

Each year the Czech-Slovak seminar “Ageing management, life management, exchange of experience” takes place, attended by representatives of all plants in both countries and where, inter alia, the issue related to cable systems is addressed. Given that, in both countries, the NPPs are of similar age and with similar cables, such information is a very important base for life assessment.

All such obtained knowledge is implemented in the AMPC. Periodic annual assessment with the outputs to the Health Reports, Safety Report is the output of the AMPC.

The AMPC in the Czech NPPs that is implemented by ČEZ, a.s. in cooperation with ÚJV Řež, a.s., won the “EPRI Nuclear Transfer Award 2016” for the “Cable Ageing Management Programme Implementation”.

At the same time, a high level of the AMPC, its implementation, maintenance and development was found during the IAEA pre-SALTO, SALTO and SALTO Follow-up missions to the Dukovany NPP [88]. In 2008, the pre-SALTO mission marked the AMPC (in particular surveillance specimens in deposits) as good practice. In 2014, the SALTO mission marked as good practice another part of the AMPC: Environmental parameters monitoring.

ÚJV Řež, a.s., plays a significant part in the implementation of the AMPC. A lot of cable qualification tests have been carried out within its accredited testing laboratory since 1994 for the NPPs and for cable manufacturers at home and abroad, including work for global major cable works such as Alcatel, Nexans or Habia Cable. At the same time, all cables of the Czech manufacturers were tested there, which are installed in the NPPs and the cables installed in the Dukovany NPP and the Temelín NPP are re-qualified on a regular basis. Experience from such, often long-term, tests is applied in the updates of the AMPC.

Quality assurance

The general principles of quality assurance for the ČEZ, a. s. are referred to in Chapter 2. Under the AMPC, the requirements for quality assurance are fulfilled by the ČEZ operating organisation without deficiencies. This is stated in the final report of the IAEA SALTO mission to the Dukovany NPP in 2014 to review the preparedness for long-term operation (LTO) of units.

Workers implementing the AMPC in the Czech NPPs hold the certificates of conformity with the requirements of ISO 9001:2008, EN ISO 14001:2004 and BS OHSAS 18001:2007. In addition, the Testing Laboratory 2305 of the ÚJV Řež, a.s., holds the Certificate of Accreditation according to the ISO/IEC 17025:2005 “Determination of selected physico-chemical, mechanical, thermodynamic and electrical properties of materials and industrial products to verify their functionality in the environment of both nuclear and non-nuclear plants; determination of parameters of radiation fields of gamma radiation and accelerated electrons” and holds a certificate of conformity with the requirements set out in the document US NRC 10 CRF, Part 50, Appendix B on the quality of work in nuclear power plants.

Ageing assessment of electrical cables of the LVR-15 nuclear research reactor

Cables of the LVR-15 research reactor are located in the environment without extraordinary influences and are only exposed to gradual, long-term, natural degradation manifested by minor deviations from the initial state.

For cables of the PCS system, new cables are qualified for the guaranteed life of the whole system, i.e. until 2030.

I&C - condition assessment of the cables of the selected I&C circuits as well as the system directly connected to the protection system was carried out under the PCS restoration project. The report "Verification of the function of the cable systems based on the measurement of selected I&C circuits for the conditions of their continued operation in the LVR-15 site, Centrum výzkumu Řež s.r.o., including emergency mode", DITI 2305/137 [89] is the output.

The qualification was based on the cable insulation resistance testing and the tension testing carried out on insulation samples from the existing cables of the selected I&C circuits and based on experience from the Ageing Management Programme for Cables (AMPC) in the Czech nuclear power plants and based on knowledge from the qualification type tests in the Dukovany NPP - for all cable types covered by this report, it is possible to document their use in the Dukovany NPP under the conditions equal to or worse than those in the LVR-15 site, Centrum výzkumu Řež s.r.o.

The cables covered by this report passed, with a large margin, the tests of electrical and mechanical properties. The condition assessment of these cables uses experience from the Dukovany NPP, where their life is monitored under the Ageing Management Programme for Cables (AMPC), which is at least 40 years under the similar conditions prevailing on the LVR-15 research reactor. For these reasons, it can be established that the cables in question have their life sufficiently qualified for their operation on the LVR-15 research reactor, Centrum výzkumu Řež s.r.o., until 2030.

Emergency power supply system (EPS1, EPS2) - New or replaced cables under the reconstruction of EPS 1 and 2; cables of the CXKE-R type, fire retardant design according to ČSN IEC 332.3, Category A. The life of these cables is reliably ensured until 2030.

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

3.1.3.1 Monitoring, testing, sampling and inspection activities for electrical cables of the Dukovany and Temelín Nuclear Power Plants

Activities under the AMPC

The basic activities under the Programme are summarised in Chapter 3.1.1.1, part "Brief description of the AMPC" and in Fig. 3.1. The actual implementation of the AMPC takes place according to the controlled documents (standards, methodologies) referred to in Chapter 2 and the time schedules for individual activities in the NPP. The main activities according to the ČEZ standards referred to in Chapter 2 are described in the following points (in more detail hereafter):

- Updating of the list of safety cables according to the SSK
- Visual inspections of cables
- Measuring of the surveillance cables

- Assessment of the cables removed from operation
- Environmental parameters monitoring
- Documenting the condition and lifetime of cables

Updating of the list of safety cables

The list of safety cables is updated in the light of data from the SSK system on an annual basis. For each type of safety cables, as much information as possible should be obtained on material composition, manufacturer's data should be obtained and accelerated ageing tests should be carried out, i.e. the trend of change in properties with a period of ageing under different conditions, etc., should be identified. Information about safety cables is then summarised in separate documents and in the database system.

Database system for safety cables

Database system for safety cables is a SW tool that is composed of three main program applications:

- Calculations and assessment of the lifetime of safety cables
- Environmental parameters monitoring
- Visual inspection reports

The system is the network application on NPP operator's computers. The application "Calculations and assessment of the life of safety cables" processes data from the Cable management system (SSK) database. Information on individual applications of the safety cables system is referred to in below.

Visual inspection of the cable routes

Visual inspections of cables provide information on the presence or absence of cable degradation and its evolution with regard to the period of operation. It is preferably carried out at the locations with potential risk of degradation (e.g. potential for small leak in case of pipework or valve failure). The visual inspection allows simple and fast detection of some cable´s degradation effects. Findings of the visual inspections are addressed by remedial actions under the maintenance programme. The significant findings include cracked jackets or cable insulations, if accessible. Remedial actions are immediately taken for the cable that should be functional in the case of design basis accident and that was diagnosed for significant deficiencies. The same stringent rules apply to the assessment of non-hermetic cable connections to a safety device that requires qualification. Other degradation indicators should be assessed individually depending on the current condition of cables, the environment and equipment to which the cable is connected.

All results of visual inspections, both positive and negative, are electronically recorded in the database. Information from visual inspections including photographs is shared within the computer network of the operating organisation. Based on the feedback set, retrospective verifications of the removal of findings are carried out.

Visual inspections were currently carried out for cables in the hermetic zone (HZ)/containment and the pipeline coridors and on all units of the Dukovany NPP as well as the Temelín NPP. In the harsh environment rooms such as Steam Generators box, pressurizer, longitudinal intermediate building, visual inspections were repeated. In addition, they were carried out in the DGS, auxiliary building and other selected locations. Visual inspections also continue without interruption in other sites.

NPP staff also carries out additional inspections under the "In-service inspection programme" (e.g. ECAD system).

Measuring of the surveillance cables (cable deposits) under the AMPC

Cable deposit means a surveillance specimen of safety cable, that is, in terms of the degradation conditions, deposited in the well defined environment of the NPP. Cable deposit is intended for periodic testing of mechanical, electrical and physico-chemical properties in order to assess the condition of cable of the same type and to determine its residual lifetime. The measurements are based on the international standards, e.g. [90], [91], [92].

There are several pieces of each type of cable deposit in the deposit that are subject to the measurement at an interval of 2 to 7 years. The measurement interval depends on the real state of cable and on the conditions of its installation. There are also cable samples in deposit that are bent over a sharp edge or pressed. The effect of possible improper cable installation is monitored on these samples. There are also backup cables that are used for the long-term preparation of defined aged safety cables. These can be used in the future, for example, for the purposes of qualifying the cables under the design basis accident conditions and post-accident state, or for the case of verifying the functionality in severe accidents.

A total of 38 cable deposits can be found in the Dukovany NPP and the Temelín NPP, where more than 400 cables are subjected to ageing. In case of newly installed types of safety cables, new surveillance samples are being prepared. The deposits with their parameters cover the cables in operation from the worst conditions in close proximity to the main circulation piping up to the mild conditions in corridors. In the Temelín NPP, cable deposits were installed before the commissioning of the plant. Fig. 3.3 shows some deposits. In the Dukovany NPP, first surveillance samples were deposited in 2005, i.e. 20 years after commissioning.

The following cables were used as surveillance samples:

- New, i.e. newly installed cables.
- Older cables from the warehouse those are adequately pre-aged.
- Cables removed from NPP operation during equipment replacements.

The results of measurement are electronically recorded in the database on an annual basis. The trends of changes in the properties of surveillance cables measured in that year are always described in the annual report of the AMPC. The reports describing surveillance cables include graphic summary of all results of the measurements of mechanical properties.

Fig. 3.3: Deposits with surveillance cables

Since 2012, additional measurements have been carried out on Dukovany NPP cables that were disconnected as part of exchange but remained in the place of original installation. Electrical parameters are measured on these cables. These parameters or trends of changes are compared to the laboratory-aged samples, with the life related to ductility of jacket and core insulations while maintaining electrical function.

NPP cables are also subject to periodic diagnostic testing in order to assess the current function of cables or the whole route. This testing is not primarily intended for the assessment of residual lifetime. The individual cable routes are tested including leads and assessed during periodic inspections of specific equipment, e.g. neutron flux detector test, thermocouple calibration, pressure transducer calibration, valve function test, etc. Periodic measurements include operational revisions of equipment including cables. All results are recorded and archived.

Assessment of the cables removed from operation

Measuring cables that were removed from operation is also part of the AMPC. The samples of cables from operation were formerly obtained randomly, for example JYTY, CYKY, CHKE-R cables in operation as part of qualification testing. The list of the cables cancelled as part of different modifications in the NPP has been available in advance since 2014. This list is compared to the current list of safety cables and if necessary, the competent equipment manager is asked for the appropriate cable for the AMPC. In 2017, the extensive project “VVK Validation and Verification” takes place, in which tens of cables will be removed from the Dukovany NPP and tested. For cables, mechanical properties will be measured and compared to other analyses.

The cables in operation are also tested during visual inspections; see details above.

Environmental parameters monitoring

The measurement of environmental parameters began in the hermetic zone areaof the Dukovany NPP in 1996, mostly in areas with expected high temperature and radiation. The measurement was gradually extended to other areas of the Dukovany NPP and the Temelín NPP. The results are used not only for the AMPC but they may be widely applicable to other NPP disciplines, typically to equipment qualification. The environmental parameters are currently measured at several hundred locations on the reactor units and at several ten locations outside the HZ/containment. If requested by operating organisation, the environmental monitoring is repeated at selected locations.

The following environmental parameters have been monitored in the Dukovany NPP since 1996 and in the Temelín NPP since 2000:

- Temperature: self-powered recorders are used with the measurement interval of 1 hour to 6 hours for a period of at least one campaign
- Moisture: measurement by means of self-powered recorders
- Radiation dose by means of alanine dosimeters with the evaluation on the spectrometer
- Measurement of thermal and fast neutron fluence rate - only in the area of the main circulation piping.

All measurements of environmental parameters in the Dukovany NPP / Temelín NPP are given in electronic sections of individual floor levels. This output is part of the SW application (Fig. 3.4).

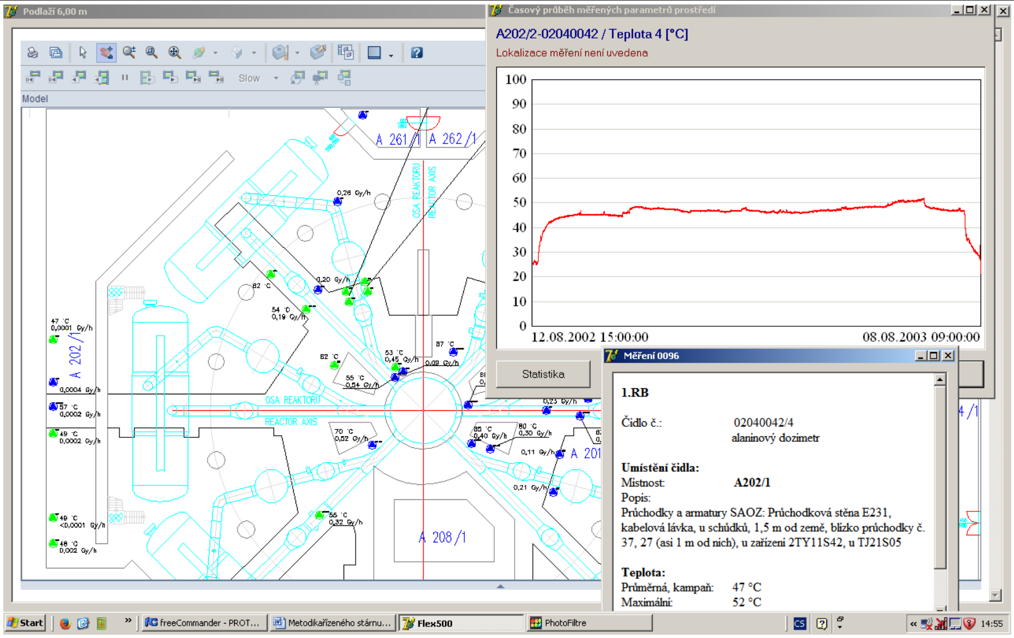

Fig. 3.4: Example of the presentation of the environmental parameters monitoring results of the Dukovany NPP and the Temelín NPP in the SW application.

<u>Residual lifetime calculation</u>

The following information is required to calculate the residual lifetime:

1. List of safety cables
   Information about cables in operation is obtained from the updated SSK databases in the Dukovany NPP and the Temelín NPP. The calculation system selects safety cables from the SSK, and assigns basic data on the overall route, input and terminal devices to them.
2. Environmental parameters
   Each room where the safety cable is located has information on temperature and dose rate. If no real environmental parameters have been measured, the design values are used. The system uses the Arrhenius equation as a prerequisite for the conversion of ageing time at different temperatures. This method is recommended in the international standards IEC, IEEE or ISO for the ageing of cable sets and in the IAEA, EPRI methodologies, etc.
3. Data from the laboratory-aged cables where the trend of changes in properties with a period of ageing under different conditions was measured (the so-called "calculation algorithms").

The system calculates separately the residual lifetime of jacket and core insulation at individual locations where the cable is in operation. It is possible to display either overall information on all calculations or summary calculation when only the worst result is displayed indicating the location.

Elongation at break of jacket and conductors insulations is the basic criterion by which the lifetime is calculated. The drop of elongation at break to the value of 50% is the criterion for the end of life. The selection of this criterion is justified in Chapter 3.1.2.1, part "Acceptance Criteria" and in the documents [78], [80], [81], [83]. The final output of the calculation is then the period that remains to achieve the critical state, the so-called "residual lifetime"; see diagram in Fig. 3.2 (page 61).

Performance and condition monitoring system for cable sets of the Dukovany NPP

The performance and condition monitoring system for cable sets in the Dukovany NPP is based, in accordance with applicable legislation, guidelines, procedures and control documents for the Maintenance and Care of Assets Planning Programme, on the implementation of the programme for plant walkdowns, in-service inspections, preventive maintenance or predictive maintenance on the basis of the results of diagnostics, functional tests and testing.

The preventive maintenance programmes are developed taking into account the safety importance of equipment (graded approach), experience with equipment operation so far and taking into account external industrial experience. Carrying out the activities under the preventive maintenance programme ensures the required availability and performance of cables sets while meeting the safety requirements.

The objective of preventive maintenance is to carry out, within the specified periods and in the specified range, activities leading to the verification and achievement of the appropriate physical condition of equipment.

The preventive and predictive maintenance programmes are reviewed on a periodic basis. The obligation to periodically review the preventive maintenance programme is imposed on the administrator of control documents for the recording, monitoring, evaluation of performance and condition, component specific ageing management of assets, technological systems and NPP equipment. The principles for evaluating and analysing the maintenance are set out in the control documents in order to develop a memory for equipment maintenance and to take such preventive actions from its analysis to ensure continuous optimisation of maintenance programmes and verification of their effectiveness in the subsequent operation of equipment.

In addition, inspections are carried out that involve a separate action or part of the inspection or revision, during which it is predominantly visually monitored whether or not the equipment meets the requirements of applicable standards and codes and does not show any apparent defects affecting or compromising the operation of the equipment itself or its vicinity.

The walkdowns, as defined in the control documents of ČEZ, a. s., ensure preventive identification of equipment failures, before the failure develops. The activities during inspection walkdowns are carried out by personnel of ČEZ, a. s., according to the walkdown checklists. The walkdown checklists do not replace the relevant operating regulations for the systems or technological units in question.

The inspection walkdowns aim in particular at:

- Inspecting the cleanliness of the environment in cable rooms and compartments
- Inspecting the room lighting conditions, emergency lighting function
- Inspecting the individual areas in terms of occupational safety and fire protection
- Inspecting the room air conditioning condition and function
- Inspecting the state of cable connection to switchboards

- Visually inspecting the cable sets

The report on walkdown contains only the activities carried out and the findings. The results of inspection and walkdown activities shall be recorded by responsible person in the (Shift Operations Logbook).

The results of periodic walkdown inspections are recorded in the form of an entry in the electronic operations logbook, identifying the type of inspection and analysing its result. The detected defects not rectified during walkdown are recorded, by default, in the defect registration card and are subsequently rectified as part of random maintenance.

Inspection history, trend monitoring. Summary of information

Inspection history in the NPP and trend monitoring are described in the chapters above. For better orientation, individual information is summarised once again.

a) *Visual inspections of cables*. Cables in operation are subjected to periodic visual inspections. The results are entered in the relevant database where the entire history is recorded. Any fundamental deficiencies are immediately rectified.Random inspections are carried out by personnel implementing the AMPC.
b) *Environmental parameters monitoring*. The monitoring of environmental parameters has been carried out for more than 20 years. The comparative measurement of temperature and dose rate was carried out at selected locations in the Dukovany NPP over the period of 3 to 13 years. Data served for obtaining information on potential trends, changes; which were not, however, confirmed.
c) *Changes in the properties of cable deposits, their degradation*. A large number of surveillance cables are installed in both plants. In the Temelín NPP, they are installed from the beginning of its operation and in the Dukovany NPP, they are installed from 2005. For these cables, their functional characteristics are gradually measured. All measured data are entered in the database.
d) *Cables removed from operation*. Cables removed from operation during replacements represent a very important item in the evaluation of the current properties of cables. After comparison to the default values, they are used for estimating other trends of monitored parameters.
e) *Preventive maintenance programmes*. The preventive maintenance programmes are reviewed on a periodic basis. History and trends are recorded and processed by equipment manager according to the approved methodologies for NPPs.

3.1.3.2 Monitoring, testing, sampling and inspection activities for electrical cables of the LVR-15 nuclear research reactor

Cable routes are subjected to periodic inspections in accordance with the In-service Inspection Programme at intervals of inspections on associated equipment. The results of inspections are recorded in protocols and stored. The major portion of cables is accessible on a daily basis; other cables are accessible for inspection during reactor outage. Any degradation is detectable at an early enough stage and allows for taking the necessary remedial actions.

3.1.4 Preventive and remedial actions for electrical cables

3.1.4.1 Preventive and remedial actions for electrical cables of the Dukovany and Temelín Nuclear Power Plants

Criteria, remedial actions and procedures related to the AMPC are described in this chapter. Basic technical acceptance criteria are referred to in Chapter 3.1.2.1. The period (in years) that remains to achieve the critical state, the so-called "residual lifetime", is the final output of the AMPC.

There are three basic categories:

1. Cable in operable condition. Residual lifetime is over 10 years.
2. Cable near the end of its life. Residual lifetime below 10 years.
3. Non-compliant cable. These cables are the cables that became non-compliant due to, for example, mechanical damage, etc.

Where cable is assessed as near the end of its life, remedial actions are applied to extend the residual lifetime. They are a set of technical measures such as improvement of operating conditions for cables, temperature load reduction with the use of an adequate barrier (Fig. 3.5), cable backing in the place of resting on the edge, etc.

Any non-compliant cable should be immediately repaired or replaced. Procedure for the replacement of cable: operator's responsible person opens a request in the SW application "TIPOM", with continuing to replace the existing, damaged cable at the end of its lifetime with the use of project tool in the form of design change. An equivalent of replacement of the original cable is fixed. Design and detail design documentation is drawn up including routing, outer connection drawings, switchboard single line diagram, according to the SSK (this database contains the actual laying of cables at the relevant location), followed by laying of a new cable according to the principles of Design amendment No. 455 "Principles of Cabling Solution (Temelín NPP)", or ČEZ_ME_0777 SSK Application - binding procedure for modifications with an impact on cables of the Dukovany NPP [93].

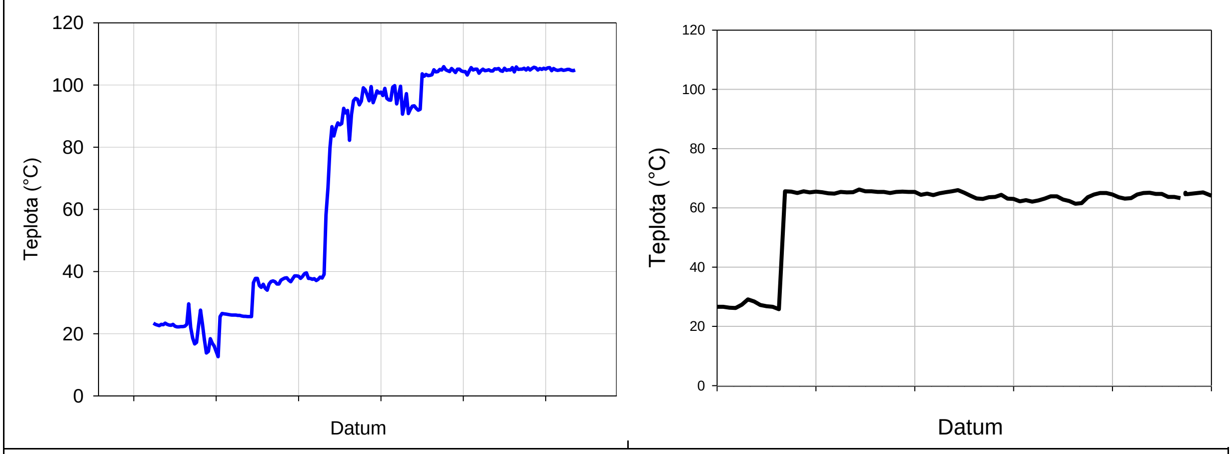

Fig. 3.5: Change in ambient temperature trend after the application of remedial action. Before application of remedial action on the left, after application of remedial action on the right

3.1.4.2 Preventive and remedial actions for electrical cables of the LVR-15 nuclear research reactor

From the perspective of the LVR-15 research reactor operator, preventive actions concerned to cables are the qualification of cables and the determination of residual lifetime that is set until 2030 for the above-described cable types. In the care of LVR-15 research reactor equipment, operating experience feedback is used from the NPP where the same cable types are installed, however, working in the environment worse than the LVR-15 location. Furthermore, cables are periodically inspected under the In-service Inspection Programme and as part of personnel walkdowns. Remedial actions involve then the replacement of cable in case of detected degradation.

3.2 Operator's experience of the implementation of the AMPC

3.2.1 Dukovany and Temelín Nuclear Power Plant operator's experience of the implementation of the AMPC

The AMPC began in 1995. With the deepening knowledge about cables, on the basis of own experience as well as experience from the world, the analyses of degradation mechanisms, condition diagnostics of insulation materials and identification of the trends in the development of degradation of insulation materials continue to extend and precise. Information about the residual lifetime of monitored cables is subsequently reviewed and periodically specified by calculation. Diagnostic of monitored cables is constantly updated with the findings obtained on the basis of taking other samples of cables from the operation, from testing of surveillance cables in deposits and from the conditions of cables identified during visual inspections. All results are used for specifying the design parameters for the assessment of residual lifetime. On the basis of the AMP results and in parallel with the ongoing qualification programme for cables, several recommendations have already been issued to replace the older types of cables from the most stressed locations, e.g. KPOBOV, KPOSG, KMPVEV cables.

For example, over 70 samples of cables taken from different locations in the Dukovany NPP, including the worst locations such as Steam Generators box, were obtained in 2016 and 2017. Their current condition was compared to the forecast provided by the AMPC. No degradation has been observed that would not be in line with the expectation according to the AMPC.

Environmental parameters monitoring, which was originally used only for the purposes of the AMPC, is currently used also for other ageing management programmes or for the specification of the life of I&C components in order to maintain NPP equipment qualification. The results of environmental monitoring serve as a basis for the LTO (Long Term Operation) of the Dukovany NPP.

The AMP for the groups of cables mentioned in Chapter 3.1.1.1, part "Brief description of the AMPC" may not yet have changed. There were no indications that the ageing of individual cable types was different from the expected process.

Amendments and additions to the AMP were gradual and always associated with the development of knowledge and needs of the NPP. The AMPC is fully functional system for the assessment of operability and knowledge of the residual lifetime of safety cables, which forms an integral part of the annual review of the types of safety cables in the HR, Periodic life assessment documents, FSAR.

The AMPC has been thoroughly examined in the international SALTO mission organised by the IAEA in 2014 and subsequently in the follow-up mission in 2016. In its conclusion [88], the SALTO mission stated that the AMPC is conducted properly and no deficiency has been identified. In addition, the mission has identified the system of cable deposits as “good performance” and has classified the environmental parameters monitoring with higher degree, as “good practice”.

Properly conducted AMPC was also appreciated by the EPRI, when the ČEZ company and ÚJV Řež, a.s., won the “EPRI Nuclear Technology Transfer Award 2016” for the “Cable ageing management program implementation”.

3.2.2 LVR-15 nuclear research reactor operator’s experience of the implementation of the AMPC

The Ageing Management Programme for the LVR-15 research reactor does not include a specific programme or action for monitoring and ageing management of cables. Cable lifetime is determined by the qualification programme; cable condition is then checked under the In-service Inspection Programme and during personnel walkdowns. Cable failure did not lead to abnormal reactor operation during all phases of reactor operation.

3.3 Regulator’s assessment and conclusions on the AMPC

3.3.1 Regulator’s assessment and conclusions on the AMPC for the Dukovany and Temelín Nuclear Power Plants

The SÚJB reviewed information concerning ageing of electrical cables that was provided for the purposes of this report by the operator of the Dukovany NPP and the Temelín NPP, together with information obtained from its assessment and inspection activities.

The condition of cables is periodically assessed by the State Office for Nuclear Safety under the review of the Final Safety Analysis Report (FSAR) updated on an annual basis. The Final Safety Analysis Report provides information from annual life assessment of the cables falling within the scope of the Ageing Management Programme for Cables. In their inspection and assessment activities, inspectors periodically assess the condition of cable sets, and the activities carried out during and outside outages are assessed and controlled (inspections, replacements, reconstructions, etc.). Last but not least, the whole system was thoroughly examined during the licensing process of the Dukovany NPP units operation after 30 years of operation (i.e. for “LTO”). In the operating experience feedback, there were minor problems relating to cable sets, e.g. degradation of cables in a much shorter time than as documented in the qualification reports; however, such events were not related to the setting of the Ageing Management Programme for Cables but to the problems with suppliers.

The Programme is set for all safety-related cables, regardless of whether they are high-voltage or low-voltage cables. A whole range of activities is carried out under the Ageing Management Programme for Cables, from cable qualification for the harsh environment, monitoring and evaluation of environmental parameters at locations where cables are installed, visual inspections of installed cables, assessment of the cables removed during technology restoration, installation of cables in deposits (the surveillance programme). The AMPC has been recognised at the international level – SALTO mission in the assessment of preparedness of the Dukovany NPP for long-term operation as well as by the EPRI (the AMPC won the award for the implementation of the Ageing Management Programme for cables in 2016).

The Ageing Management Programme for Cables meets the requirements set out in applicable legislation and other documents within the scope of national legislative and regulatory framework of the Czech Republic (see Chapter 2.1).

For the reasons set out above, the State Office for Nuclear Safety considers the Ageing Management Programme for Cables set for the Dukovany and Temelín NPPs to be properly set and sufficiently effective.

3.3.2 Regulator's assessment and conclusions on the AMPC for the LVR-15 Nuclear Research Reactor

The condition of safety-related cables of the LVR-15 research reactor is not currently monitored in terms of ageing effects; the safety cables of the LVR-15 research reactor are not included in the Ageing Management Programme for the LVR-15 research reactor. Cables are monitored under the In-service Inspection Programme concerning a particular technological equipment / system in terms of their functionality. In view of the new legislation issued, the Ageing Management Programme for the LVR-15 research reactor will be adapted to the new legislation in the Czech Republic by the end of 2018 (transitional provisions). The SUJB requires that the safety cables will be included in the Ageing Management Programme for reasons of monitoring the ageing effects. The whole Programme will be reviewed by the State Office for Nuclear Safety following the transitional provisions.

4. Concealed pipework

4.1 Description of ageing management programmes for concealed pipework

4.1.1 Scope of ageing management for concealed pipework

4.1.1.1 Scope of ageing management for concealed pipework of the Dukovany and Temelín Nuclear Power Plants

The following types of concealed (or inaccessible for inspection) pipework are installed in the Czech NPPs:

- Steel pipework
  - Buried in soil
    - Insulated from inside and from outside
    - Insulated from outside only
  - Embedded in concrete block below ground level (uninsulated)
  - Embedded in concrete in buildings (e.g. spent fuel pool cooling system, penetrations through vertical and horizontal supporting structures etc.)
- Polyethylene pipework (most fire water pipe sections)

The following programmes are applied to the types of pipework above:

- AMP for inaccessible (buried) pipework described in the document ČEZ_ME_1036 [94], which is focused on pipelines of the circuits for circulation cooling water, essential service water (ESW), non-essential service water (NESW), fire water and raw water that are buried or inaccessible for inspection outside the structures
- AMP for service water pipework in the document ČEZ_ME_1043 [95], which is applied to pipework of the pipeline systems for circulation cooling water, essential service water (ESW), non-essential service water (NESW), fire water and raw water in buildings. However, this Programme is not primarily focused on ageing management for inaccessible sections of this pipework; the condition of inaccessible sections of these systems inside buildings is predicted on the basis of knowledge of the condition of pipe sections for inspection of accessible pipe sections, visual inspections and repair history
- The maintenance programme (in this chapter, in terms of the range of other pipework systems that are not included in the above mentioned programmes (such as spent fuel pool cooling system pipework that is partially embedded in concrete inside buildings and inaccessible pipe sections of other systems – penetrations through structures, etc.))

There are no buried pipelines in the Dukovany and Temelín NPPs that contain radioactive effluents or diesel generator fuel lines.

The Ageing Management Programme for Concealed Pipework [94] was developed by the operator of the Czech NPPs in 2016. The Programme is focused on concealed pipework or pipework inaccessible for inspection, located outside the buildings. Before 2016, such pipework was only subject to routine maintenance because most of the underground or concealed pipework was not considered as relevant to operational reliability and to safety due to its redundancy and existing operating experience. However, increased attention has been given to these types of pipework in recent years. In the Czech NPPs, among others following the event in the Dukovany NPP in 2014, which involved a large leak from a part of buried ESW pipework, the operator decided to implement the Ageing Management Programme for Concealed Pipework in order to get a better insight into the condition of these lines that are difficult to inspect by direct methods. This Programme, with regard to the date of its development, is gradually implemented and it will be revised on the basis of the assessment of its effectiveness.

The Programme is based on corrosion degradation risk assessment of individual lines by means of the EPRI BPWORKS™ application, which is supplemented with periodic inspections and condition assessment of the outer insulation of pipework. In addition, some sections of buried pipework are newly instrumented for the EDMET (electrodiagnostics of magnetic pipes) measurement method which allows specifying the mean wall thickness for the sections measured. Ultrasonic thickness measurement is carried out at accessible locations in buildings (turbine buildings, pump stations, gravity water reservoir) as well as in sumps. Special tightness tests have been carried out for water supply systems since 2016. From May 2017, the Programme includes recording of maintenance actions, stating the reason for repair, causes of damage, corrosion attack and wall thickness of original pipework before repair. Some of these parameters then enter the BPWORKS™ providing more accurate risk estimate.

The Ageing Management Programme for Concealed Pipework [94] is complementary to the AMP for service water pipework [95], which addresses the issue of ageing of pipework of the same systems – i.e. circulation cooling water, ESW, NESW, fire and raw water but, unlike the Programme [94], it is focused on pipelines inside buildings. The issue of ageing of other inaccessible pipelines that are not directly included in the mentioned AMPs (e.g. pipework for other systems such as those embedded in concrete inside buildings) is currently addressed through maintenance activities, which, in addition to other activities, involve daily walkdowns in which the surface condition of pipework and the surface condition of ceiling and floor walls are recorded in the Shift Operations Logbook (rust stains, extracts, deposits and potential leaks). The AMP for service water pipework requires a periodic evaluation of of any negative trends in these records. A gradual extension of these two AMPs is expected on the basis of implementation experience and evaluation of a feedback. Information from the EPRI shows that it will become possible to assess pipework inside buildings in the BPWORKS™ application from 2018; incorporation of such extension of the BPWORKS™ application in the AMP for service water pipework is scheduled for 2020.

In view of the scope of activities and measured parameters, the text hereafter concerns mainly the Ageing Management Programme for Concealed Pipework [94].

Line scope-setting for the Concealed Pipework AMP

The Ageing Management Programme for Concealed Pipework [94] includes safety relevant systems, i.e. ESW systems and fire water piping as well as systems important to unit(s) operability, e.g. circulation cooling water systems, NESW systems and water supply systems. It primarily includes all pipes below ground level including pipework in concrete block, and does not include pipework in accessible or non-accesible channels and pipework inside buildings. The scope of pipework included in the Programme is broader than the scope arising from general rules for selecting (screening) equipment within the scope of the ageing management, as described in Chapter 2.3.1.1. Welds, as an integral part of pipe sections, are covered by the Programme. In the Dukovany and Temelín NPPs, flanges are not buried and are not therefore included in the Programme. The advantage of the broader concept of equipment included within the scope of the Programme is that in case of obtaining more detailed information about safety irrelevant section, it allows for the transfer of information about degradation of “non-safety” lines to safety lines of similar construction and vice versa, which contributes significantly to improving knowledge of the condition of these pipework systems.

The AMP includes pipework of the following systems:

Raw water – Dukovany, Temelín

Raw water lines (Water Supply Systems) in both NPPs are buried and are protected with an asphalt layer from the outside. These lines are additionally protected from the inside with an asphalt layer in the Dukovany NPP, and with epoxide coat in Temelín NPP.

Circulation cooling water – Dukovany, Temelín

Most circulation cooling water lines in both NPPs are encased in concrete block and only short sections before the Central Pumping Stations I and II in the Dukovany NPP are buried and protected with an asphalt layer from the outside. In the Temelín NPP, the circulation cooling water line is embedded in concrete block.

Essential service water – Dukovany, Temelín

Most essential service water lines in Dukovany NPP are embedded in concrete block. Short sections in front of Central Pump Stations I and II in the Dukovany NPP are buried and protected with an asphalt layer from the outside. In the Dukovany NPP, these buried portions of ESW are currently replaced by new lines, made from the same material, with increased pipe thickness from 8 to 10 mm. In the Temelín NPP, ESW lines are installed in accessible pipe tunnels.

Non-essential service water – Dukovany, Temelín

Most non-essential service water lines in both NPPs are embedded in concrete block and only short sections in front of Central Pump Stations I and II in the Dukovany NPP are buried and protected with an asphalt layer from the outside. In the Temelín NPP, NESW feed line is embedded in concrete block in parallel with circulation cooling water. NESW is brought together with circulation cooling water in the turbine building and there is no NESW return line in the Temelín NPP.

Make-up water - Dukovany

In the Dukovany NPP, make-up water lines for circulation cooling water circuits are buried.

Identification of ageing mechanisms related to concealed pipework

Degradation mechanisms, attacking pipework within the scope of the above mentioned Programmes, were assessed in the “Ageing Management Review” using of the Catalogue of Degradation Mechanisms, identifying and assessing potential and real degradation mechanisms. The BPIG (Buried Pipe Integrity Group) and the BPIRD (Buried Pipe Inspection Result Database) are another important source summarising operating experience of the NPPs participating in the EPRI. The process for the identification of possible degradation mechanisms is described in detail in Chapter 2. Corrosion is the main degradation mechanism for concealed pipework including welds, which form an integral part of pipework.

4.1.1.2 Scope of ageing management for concealed pipework of the LVR-15 nuclear research reactor

For the LVR-15 research reactor, cooling system and ventilation system of the Radiation Controlled Area could be considered as safety relevant concealed pipe sections.

The cooling system has the concealed pipe sections within second cooling circuit. In the ventilation system, concealed pipe section is a duct of controlled exhaust from the reactor hall.

Secondary circuit

The secondary circuit connects primary exchangers located in the reactor pump building (building 211) and secondary exchangers located, together with the pumps, in the water treatment building. A distance between buildings is approximately 100 m.

Two steel pipelines of the second cooling circuit DN 500 mm (delivery and return) are installed between the reactor building and the water treatment building. Pipes are insulated with the coal tar enamel and buried at an approximate depth of 2 m.

The cooling circuit is closed, filled up with technical water in the water treatment plant. Water volume in the circuit is approximately 65 m^3. For safety reasons, operating pressure is higher comparing to primary coolant system and it is at least 0.45 MPa. Pressure in the system is accomplished by filling up through a pressure reducing valve. The approximate flow of 800 m^3/hour is normally maintained in the circuit. Temperatures of up to 38°C are normally at reactor building outlet and water cooled down to temperature of approximately 30°C returns to the reactor from the water treatment plant.

Process ventilation system of the LVR-15 research reactor

The system is intended for disposal of radioactive gases, produced during reactor operation, for maintaining constant underpressure in the reactor, pump room and hot cells, and for air exchange in the main reactor hall. Compliance with the above mentioned requirements is based on the principle of underpressure ventilation.

Vacuum ventilation duct is made from carbon steel. Approximately 80 % of the length is buried below ground. In this part, it is insulated with the coal tar enamel. None of these lines are included within the scope of the Ageing Management Programme for the LVR-15 research reactor.

4.1.2 Ageing assessment of concealed pipework

4.1.2.1 Ageing assessment of the Dukovany and Temelín Nuclear Power Plants concealed pipework

The assessment of concealed lines is carried out under the AMP for concealed pipework [94] focused on the monitoring of a risk of corrosion degradation, which is the most significant degradation mechanism, and by monitoring the specified parameters, on the basis of which the extent of corrosion damage is identified.

For periodic monitoring of lines, the operator of the Dukovany and Temelín NPPs developed a new EDMET method for the measurement of mean residual wall thickness in the framework of its research activities The method is based on the measurement of electrical resistance together with reflectometry measuring an electric signal propagation speed between the conductor (pipe wall) and the insulator (dielectric, outer pipe insulation). The reflection is generated in a location of relative permeability change (i.e. moisture between pipe insulation and metal), which can indicate insulation damage. Based on the electric signal propagation speed and the reflection time, it is possible to very precisely calculate the distance of first moist location from the electrode with an accuracy of centimetres. Reflectometric measurements have been verified under the laboratory conditions and also once in the plant. The reflectometry was successfully used for measuring a length of uninsulated pipe from the electrode up to the place where the pipe is brought beneath the water level. Measuring points are installed on newly repaired ESW lines, and zero and first repeated measurements are carried out.

The first assessed parameter is the final risk taken from the EPRI BPWORKS™ application. This risk has values ranging between A and C. The EPRI BPWORKS™ is the software developed by EPRI in order to assess the risk of corrosion damage to buried pipelines. Design parameters of the lines in the NPPs, properties of protective layers, soil properties and properties of medium inside lines as well as consequences of line leak, rupture and occlusion are entered in the application database. Subsequently, the application uses the data for point and colour assessment of the risk of the leaks initiated from outer and inner side of piping, the risk of the rupture initiated from inside and from outside, and occlusion. For the purposes of AMP, the line score is equal to the score of its worst section. The degradation risk assessment performed by the application also depends on a quality of input data (information about pipelines); the software can modify the final risk on the basis of the results of inspections on particular line (e.g. EDMET method, planned inspections combined with excavation work and random inspections during unplanned excavation work, etc.) and similar lines. The assessment in colour allows a targeting of next inspections and activities associated with buried pipework, which can be directed towards a group of lines with similar properties.

Other parameters to be assessed under the [94] are as follows:

- Tightness check of the raw water supply lines (average leak during measurement is monitored [l/hour], measurement time [hour] and any leak is localised)
- Minimum measured wall thickness [mm]
- Condition of the insulation of water supply lines with the use of the DCVG method (electrical conductivity of defects, soil resistivity, natural electrical potential of pipework
- The results of EDMET measurements (ESW testing near the Central Pump Station) – mean wall thickness [mm], insulation failure localisation
- Corrosion parameters
- On-line acoustic measurement – number of recorded leaks
- Results of inspections of fire water lines
- Repairs and replacements (number, reason)

Acceptance criteria

The BPWORKS™ application itself does not contain any own acceptance criteria except for minimum thickness, which is entered as input value in the database. Determination of the criterion thickness is based on the normative technical documentation “Evaluation of the Equipment and Pipework Strength for VVER Nuclear Power Plants”, NTD of the Association of Mechanical Engineers Section III [96] and standard ČSN EN 13480-3 [97]. The specific method of calculation is described including spreadsheet calculation sheet in Annex to the AMP “Determination of the criterion thickness for buried pipework”.

Internal and external operating experience

Degradation mechanisms attacking buried pipework have been identified on the basis of operating experience from the Dukovany and Temelín NPPs. Furthermore, operating experience is periodically shared between ČEZ and SE departments. This exchange of experience provides next informations on other degradation mechanisms of pipelines from the Jaslovské Bohunice and Mochovce NPPs. Experience from Slovakia shows that open-circuit pipework in concrete blocks, which was exposed at random, shows hardly any damage after approximately 30 years of operation. In addition, corrosion attack was confirmed under damaged insulation of buried pipes, as in the Czech Republic. The BPIG and the BPIRD are another important source summarising operating experience of the NPPs participating in the EPRI.

4.1.2.1 Ageing assessment for concealed pipework of the LVR-15 nuclear research reactor

The operating conditions (temperature, overpressure) of these pipes are very low; at the same time, safety analyses (carried out during stress tests following the accident in the Fukushima Daiichi NPP) demonstrated very low impact of potential failure not resulting in fuel damage or release of radioactive material into environment. For these reasons, no periodic inspections on buried pipework are planned under the ageing management of the LVR-15 research reactor; intermediate inspections are carried out on the occasion of an exposure of the relevant pipelines for other reasons, e.g. during construction work in nearby locality.

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

4.1.3.1 Monitoring, testing, sampling and inspection activities for concealed pipework of the Dukovany and Temelín Nuclear Power Plants

Inspections on buried pipelines are difficult because most inspection techniques require extensive excavation work. The inspection activities carried out on the pipework are as follows:

- Ultrasonic thickness measurement and visual inspection from the outer surface - they are carried out whenever the pipeline is exposed, for some reason, even not associated with the pipeline
- Visual inspection from the inner side - it is carried out on circulation cooling water line in the Dukovany NPP when draining the circuit
- Ultrasonic residual pipe thickness measurement at accessible locations in sumps and in the Central Pump Station on the lines that are buried down- or upstream; the measurement is carried out according to the place of inspection at an interval of 6 or 8 years
- Aerial thermography to identify locations with raw water supply line leaks outside the plant site
- Measurement of the direct voltage potential drop for raw water supply lines allowing for the identification of the location of damaged asphalt insulation (Condition monitoring of the insulation of water supply lines with the use of the DCVG method) – the measurement is carried out every four years; the worst identified parameters of the pipeline in question are entered in the assessment.

  The following is measured:
  - Electrical conductivity of defect (A – small defects: 0-15 % IR, which can remain unrepaired provided that functional cathodic protection is present in the area of defect on the pipe, B – medium defects: 16-35 IR, which are periodically monitored, C – large defects: 36 – 70% IR – repair planning is

required; they are big protective current consumers in case of cathodic protection and are significantly endangered by corrosion in the absence of cathodic protection, D – serious defects: 71-100% IR, which should be immediately repaired. IR is the percentage potential difference with the cathodic protection activated and deactivated in the are of defect on the pipe

  - Soil resistivity – soil aggressivity is measured in [Ωm] (A – very low: >100 Ωm, B – medium: 50 to 100 Ωm, C – increased: 23 to 49 Ωm, D – very high: 0 to 22 Ωm) – additional parameter
  - Natural electrical potential of pipework – additional parameter (unfavourable is anodic, which indicates the potential for active corrosion processes on pipework

- Tightness check of water supply lines in the Dukovany NPP (in order to verify the tightness of raw water pipeline (raw water delivery lines from the raw water Pump station Jihlava to the gravity water reservoir). Before measurement, the delivery line is blinded on discharge side of the pump in the Pump station Jihlava on one side and in the gravity water tank on the other side. During measurement, the detached line is filled with water and any drop in water level is observed in the transparent cylinder connected to that delivery line in the gravity water tank. At the same time, water pressure is observed in the line at the Pump station Jihlava outlet to localise the potential leak of the line tested. When the pipe is leaking, the water level will drop simultaneously with the pressure. Pressure drop stops when the water level in the pipe stops under the leak. The measurement time is limited to 5 days. Average leak during measurement is measured in [l/hour], measurement time [hour] and any leak is localised. It is carried out at intervals: zero measurement, first measurement within one year after the zero measurement and following measurements every four years
- In the Dukovany and Temelín NPPs, loops with corrosion coupons are installed, which are periodically assessed according to the plan in the KOS (Dukovany NPP) and KOROZE (Temelín NPP) computer applications. The evaluation period is 28 or 56 days depending on the type of steel
- Water corrosion parameters – larger changes in corrosion parameters require implementation of any remedial actions; inspection period once a year; water corrosion parameters are monitored in the KOS (Dukovany NPP) and KOROZE (Temelín NPP) applications.
- Certain lines, which were recently repaired in the Dukovany NPP (ESW near the Central Pump Station), are equipped with the EDMET instrumentation (the method is permanently applied to buried pipes for ESW I, II and III between the Central Pump Station and the concrete block. It can also be applied to accessible pipe sections for mean thickness measurement if there is no galvanic connection between the measuring electrodes and the ground or other pipes (without supports, branch pipes, valves, hangers, etc.)). First evaluation takes place within one year after installation and according to results next evaluation takes place in one, three or five years. The measurement covers mean wall thickness, which is also entered as specified parameter in the BPWORKS™ application, and insulation failure localisation

- On-line acoustic measurement – number of recorded leaks - the method is permanently applied to buried pipes for ESW I, II and III between the Central Pump Station and the concrete block (only in the Dukovany NPP) as an autonomous monitoring system in order to immediately detect any leak that is indicated by acoustic signal emission (hissing sound of leaking water). Information about any leak detected is automatically transmitted to the operator; evaluated under the AMP once a year
- Fire water line tightness test and functionality (flow capacity) test according to the operating regulations PP 095j [98] and P132j [99] in accordance with Ministry of the Interior Decree No. 246/2001 Coll., laying down the conditions of fire safety and state fire supervision (Decree on Fire Prevention) [100]. The output is the operability inspection report; evaluated under the AMP once a year
- Repairs, replacements – systematic recording of maintenance actions, stating the reason for repair, causes of damage, corrosion attack and wall thickness of original pipework before repair – evaluated under the AMP once a year
- Additional inspections on inaccessible pipework (they are not applied permanently; they serve to clarify the operational knowledge of the condition of pipework and subsurface):
    - Thermography – aerial multi-spectral monitoring to detect or confirm any larger water leaks indicated by change in temperature field or by change in vegetation colour.
    - Ground-penetrating radar – changes in subsurface, moisture, displacements and caverns are detected by means of electromagnetic waves.
    - Metal magnetic memory method – the method uses a magnetogram to identify any changes in natural magnetic field over the length of pipe (material volume changes, thickness, welds, mechanical stress, etc.). The disadvantage of this method is sensitivity on a depth of line installation, affecting accuracy of measurement.

Establishment of acceptance criteria for any of the above mentioned methods is described in Chapter 4.1.2.1.

4.1.3.2 Monitoring, testing, sampling and inspection activities for concealed pipework of the LVR-15 nuclear research reactor

Periodic inspections on buried pipework are not implemented under the Ageing Management Programme for the LVR-15 research reactor or the In-service Inspection Programme; inspections are carried out on the occasion of an exposure of the relevant pipelines for other reasons, e.g. during construction work in the vicinity.

4.1.4 Preventive and remedial actions for concealed pipework

4.1.4.1 Preventive and remedial actions for concealed pipework of the Dukovany and Temelín Nuclear Power Plants

The most significant action applied to the Dukovany and Temelín NPPs is to maintain the chemistry including monitoring of the maximum cycles of concentration. No specific chemical criterion enters the AMP.

The risk of corrosion damage to lines identified by the BPWORKS™ application and the results of other inspections are periodically evaluated under the Ageing Management Programme [94]; the evaluation is the input to the Health Reports.

Remedial actions include analysis for individual segments where the input parameters are analysed and subsequently, it is decided on further solution depending on the particular line. The solution may include, for example, complementation by data from documentation, specification or identification of medium parameters and soil parameters, indirect inspections, direct inspections. The objective of the implementation of remedial actions is to make every effort to reduce the risk score and to classify to a lower category. In case of leak, the remedial action involves repair or replacement of the section concerned.

4.1.4.2 Preventive and remedial actions for concealed pipework of the LVR-15 nuclear research reactor

Given that the condition of concealed pipe sections is inspected occasionally, any remedial actions would be taken only in the event of deterioration of the condition of such pipe sections.

4.2 Operator's experience of the implementation of the AMP for concealed pipework

4.2.1 Dukovany and Temelín Nuclear Power Plant operator's experience of the implementation of the AMP for concealed pipework

The Ageing Management Programme for concealed pipework was put into practice last year and does not yet provide any extensive experience. The programme concept has already proven itself. In 2014, a pilot project was implemented where the selected typical buried lines were assessed with the use of the previous version of the BPWORKS™ application. Subsequent operating experience confirmed the outputs of assessment.

4.2.2 Nuclear research reactor operator's experience of the implementation of the AMP for concealed pipework

As described in 4.1.1.2, the condition of concealed pipework is not subject to periodic inspections. In the construction of the new Experimental Hall 211/12, adjacent to the LVR-15 research reactor hall, the exhaust pipe of the hall was partially exposed during construction work between 2012 and 2013, and pipe condition inspection was carried out. Under the asphalt coating, the pipe was found in satisfactory condition, with only surface corrosion and minimum residual wall thickness of 80% of the original condition (after 55 years of operation). These findings confirmed expected lifetime of the pipework is 20 years at minimum.

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

4.3.1 Regulator’s assessment and conclusions on ageing management of concealed pipework for the Dukovany and Temelín Nuclear Power Plants

The State Office for Nuclear Safety imposed, by way of the condition of the Decisions on Operation of the Dukovany NPP (“LTO”), an obligation on the operator of the Dukovany NPP to put into practice the methodology for monitoring physical condition of the ESW system (including inaccessible pipework). The frequency and scope of the inspections should be set so as to reveal, sufficiently in advance, any irregularities and defects caused by system operation, thus preventing significant malfunctions of that system. By fulfilling this condition of the decision, the Ageing Management Programme for Concealed Pipework [94] and the Ageing Management Programme for Service Waters [95] were implemented and the In-service Inspection Programme was adjusted by the operator in 2016. Implementation of these programmes was preceded by research projects or activities, e.g. development of the EDMET method, cooperation with the EPRI in the implementation of the BPWORKS™ application. From the perspective of the State Office for Nuclear Safety, both Programmes formally meet the attributes required by legislation of the Czech Republic; nevertheless, due to the recent date of their implementation, conclusions cannot be draw yet on their effectiveness. The State Office for Nuclear Safety monitors the activities carried out under that programme in the framework of its assessment and inspection activities.

4.3.2 Regulator’s assessment and conclusions on ageing management of concealed pipework for the LVR-15 nuclear research reactor

Concealed pipework of the LVR-15 research reactor is not at this stage included within the scope of the Ageing Management Programme for the LVR-15 research reactor. In view of the new Atomic Act, the Ageing Management Programme for the LVR-15 research reactor will be adapted to the new legislation by the end of transition period – i.e. by the end of 2018; the State Office for Nuclear Safety will then review the fulfilment of the requirements of new Atomic Act. However, given the scope and parameters of the medium of such pipework (application of the principle of graded approach), the State Office for Nuclear Safety does not envisage the extension of activities on such pipework even at the end of transition period of the new Atomic Act.

5. Reactor pressure vessels

This chapter describes the Ageing Management Programme for reactor pressure vessels of the Dukovany and Temelín NPPs and for reactor non-pressure vessel of the LVR-15 nuclear research reactor, which is described beyond the requirements that were set out in Technical Specifications [1] because it is one of the most important components of that nuclear installation. The titles of chapter and individual subchapters containing the term "pressure vessel", i.e. for passages concerning the LVR-15 nuclear research reactor, are not perfectly correct; however, the authors of the report followed the structure and the titles of chapters according to the Technical Specification [1].

5.1 Description of ageing management programmes for RPVs

5.1.1 Scope of ageing management for RPVs

5.1.1.1 Scope of ageing management for reactor pressure vessels of the Dukovany and Temelín Nuclear Power Plants

Reactor pressure vessels of the Dukovany and Temelín NPPs consist of a body (cylindrical vessel with an elliptical bottom head), head and components of the main flange. They are part of the reactor coolant system and perform the following safety functions:

- Maintaining integrity of the main pressure boundary of reactor coolant
- Maintaining sufficient coolant amount for core cooling in normal and abnormal operation
- Maintaining sufficient coolant amount for core cooling during and after accident conditions, under which there was no failure of integrity of reactor coolant system

To asure the above mentioned system functions the important function of the pressure vessel is integrity.

Reactor components, comprising the pressure boundary of the primary circuit, are the selected components classified as Safety Class 1, other reactor components fall into the Safety Class 2. According to the [34], the reactor is assigned criticality 1 and function important to nuclear safety according to [35] is also of category 1.

Types of reactor pressure vessels in the Dukovany and Temelín NPPs

<u>VVER 440/213 reactor pressure vessel (Dukovany NPP)</u>

The reactor pressure vessel body is cylindrical vessel, welded of one long smooth, two short smooth forged rings, two nozzle rings, flange ring and elliptical bottom head.

The upper part of the pressure vessel body consists of flange ring with an outer diameter of 4270 mm and an inner diameter of 3340 mm. There are 60 threaded holes M 140 × 6 on the front face of that ring for studs of the main flange and two pairs of grooves for nickel seal. There is a lug on the inner surface of the flange ring of pressure vessel body, which is used for positioning the reactor barrel.

Under the flange ring, there is the upper nozzles section with a height of 1400 mm, with six nozzles DN 500 for reactor coolant outlet, with two nozzles DN 250 of the emergency core cooling system and one nozzle of the instrumentation and control system.

Under the upper section of nozzles, there is the lower nozzles section with a height of 1725 mm, with six nozzles DN 500 for reactor coolant inlet, two nozzles DN 250 of the emergency core cooling system and one support collar, which is located under the row of nozzles. The pressure vessel body with the support collar fits on the support, which is mounted on the support frame of concrete reactor cavity. Recesses for keys are on the circumference of the support collar. The inner surface of pressure vessel is fitted with a separating ring, three vertical segmental baffles and eight consoles for fixation of core barrel. The separating ring is located between upper and lower nozzle sections. It fits tightly to the reactor core barrel and separates the incoming and outcoming coolant and distributes the flow through reactor.

Onto the lower nozzle ring, a long smooth ring with a height of 2700 mm is welded, onto which a short ring with a height of 1830 mm is welded, into which the core barrel centering consoles are welded. Second short ring with a height of 1895 mm is welded onto that ring and the assembly is enclosed with an elliptical bottom head. The thickness of smooth rings is 140 mm and of elliptical bottom head 160 mm.

The height of pressure vessel is 11805 mm.

Coolant baffles are located at the lower nozzles DN 250. Thin-walled bushings are mounted in the nozzles DN 250, which provide thermal protection to nozzle material during operation of the emergency core cooling system delivering of cooling water to the reactor.

Reactor internals are placed in pressure vessel. The vessel is connected to the main circulation piping loops with six inlet and six outlet nozzles DN 500. The reactor pressure vessel is connected to the lines of emergency systems with four nozzles DN 250.

Nickel seal ensures leak-tightness of the main flange between the pressure vessel and the reactor vessel upper head.

The vessel upper head is welded of the top plate and the ring, and contains the floating flange with the holes drilled on the circumference for bolts M 140 to ensure tightness of the main nickel seal and threaded holes M85x6 for bushings M 85 and pressure bolts M 64x4 to ensure tightness of the spare inner seal.

The vessel is made from chromium-molybdenum-vanadium steel; the inner surface of pressure vessel body and head is covered with a double-layer austenitic stainless steel liner. The materials used for individual RPV components are listed in the following tables.

Table 5.1: Vessel materials

Material	Component
15CH2MFA	flange ring, upper and lower nozzle ring, smooth ring long, smooth ring short, bottom, partition ring, I&C nozzle, 1st part of main circulation piping nozzle, 1st part of ECCS nozzle
08CH18N10T	2nd part of main circulation piping nozzle, 2nd part of ECCS nozzle, bracket for cavity guide
Sv10CHMFT	RPV circumferential welds (except for nozzles)
Sv07CH25N13	RPV cladding 1st layer
Sv08CH19N10G2B	RPV cladding 2nd layer
Sv-04Ch19N11M3	I&C nozzle end cap welding, weld 51/a
EA-400/10T	ECCS nozzle shroud welding (cold and hot parts), bracket for cavity guide welding

Table 5.2: Vessel head materials

Material	Component
18CH2MFA	vessel head top plate
Sv07CH25N13	vessel head cladding 1st layer
Sv08CH19N10G2B	vessel head cladding 2nd layer
22K	CRDM nozzles, NFM(EV) nozzles, TM nozzles, support rod bushing, CRDM nozzle flange, CRDM nozzle thread M36
38CHN3MFA	Stud M140
25CH1MF	nuts, washers
25CH3MFA	floating flange
08CH18N10T	CRDM nozzle shroud, CRDM nozzle flange insert, TM and NFM nozzle insert
EA-395/9	cladding on the outer surface of the vessel head under TM nozzle welding, cladding on the end face of TM nozzle
ZIO-8	CRDM nozzle welding to the vessel head (inner surface), CRDM nozzle shroud welding to the inner surface of the vessel head, top welding of nozzle shroud to TM/NFM nozzle cladding,
EA-400/10T	TM/NFM nozzle shroud (insert) welding to vessel head cladding (inner surface)

Table 5.3: Main flange materials

Material	Component
15CH2MFA	pressure ring inner, RPV flange, internal foot
25CH3MFA	floating flange
22K	compensating foot
25CH1MF	pressure bolts, pressure bolts bushing
18CH2MFA	Vessel head
Sv07CH25N13	vessel head cladding 1st layer
Sv08CH19N10G2B	vessel head cladding 2nd layer
Sv07CH25N13	RPV cladding 1st layer
Sv08CH19N10G2B	RPV cladding 2nd layer
38CHN3MFA	Stud M140
12CH1MF	expansion compensating pipe
UONI-13/55	fillet welds below the expansion compensating pipe

Schematic representation of VVER 440/213 reactor pressure vessel is shown in Fig. A.5 – A.8 in Annex A hereto.

VVER 1000/320 reactor pressure vessel (Temelín NPP)

The reactor pressure vessel is a vertical high-pressure vessel with the main flange with eight nozzles DN 850 for coolant inlet and outlet. The reactor pressure vessel is welded with circumferential welds by automatic submerged arc welding of seven parts: 3 forged smooth rings, 2 nozzle rings, one flange ring and pressed bottom head. Nozzle rings are thick-walled forged pieces; nozzles DN 850 are fabricated by hot extracting.

In the plane of upper nozzles, there are two nozzles DN 270 of the emergency cooling system and one nozzle DN 250 for outputs of in-core measurements. In the lower plane of nozzles, there are two nozzles DN 270 of the emergency cooling system in addition to four main nozzles.

There is a lug on the inner surface of the flange ring, which is used for hanging up the reactor barrel. At the bottom of the pressure vessel, there are eight weld consoles with guide keys, which prevent tangential movement of the core barrel. A separating ring is welded between the upper and lower rows of nozzles, which is during operation in contact with the core barrel, thus separating coolant inlet and outlet. The bottom part of the core barrel consists of perforated elliptical bottom with built-in supports for fuel elements. The core shroud (reactor core outer shell) is placed inside the core barrel and encloses the reactor core. The protective tubes block is placed above the core, which prevents the assemblies from floating, secures their relative positions and performs the function of control rod protection.

The vessel upper head is welded of elliptical bottom and cylindrical flange ring. The vessel head has 91 welded nozzles, used for the connection of control rod drive bushings and for outputs of temperature and neutron flux measurements. In addition, there are six bushings for support rods of steel structure of the upper block welded on the vessel head. The venting nozzle is used for venting during reactor filling with coolant.

The reactor upper block is mounted on pressure vessel body flange, which consists of pressure vessel head, linear stepping drives, steel structure of the reactor upper block, and cross beam.

The pressure vessel is mounted on the carrier ring of the support frame of concrete reactor cavity. The position of pressure vessel in the concrete cavity during earthquake or failure of the main circulation piping is secured by the carrier ring, positioned from the outside of the main flange.

The pressure vessel is made from chromium-molybdenum-vanadium steel. The inner surface of pressure vessel body and upper head is covered with a double-layer austenitic stainless steel liner.

The main construction materials of reactor pressure vessel are listed in the following tables.

Table 5.4: Reactor vessel material

Material	Component
15CH2NMFA	flange, nozzle rings, bottom, partition ring, main circulation piping nozzle, ECCS, I&C nozzles
15CH2NMFAA	support ring, smooth upper ring, smooth lower ring
Sv12CH2N2MA	RPV circumferential welds, except for welds 3 and 4
Sv12CH2N2MAA	RPV circumferential welds 3 and 4 in the area of reactor core
Sv07CH25N13	RPV cladding 1st layer
Sv04CH20N10G2B	RPV cladding 2nd layer (anti-corrosive)
08CH18N10T	containers for surveillance specimens, brackets for cavity guide, centering keys
Sv-04Ch19N11M3	I&C nozzle end cap welding
EA-400/10T	bracket for cavity guide welding

Table 5.5: Vessel head materials

Material	Component
15CH2NMFA	vessel head top plate, flange ring, floating flange
22K (carbon)	vessel head nozzles, air tank nozzle, support rod bushing
38CHN3MFA	stud M 170, nuts, washers
Sv07CH25N13	cladding 1st layer
Sv04CH20N10G2B	cladding 2nd layer (anti-corrosive)
ZIO-8	CRDM, TM/NFM nozzle welding to the vessel head (inner surface),
EA-400/10T	CRDM, TM/NFM nozzle shroud welding to the inner surface of the vessel head, cladding covering the CRDM, TM/NFM nozzles (inner surface)

Table 5.6: Main flange materials

Material	Component
15CH2NMFA	Vessel head
38CHN3MFA	Stud M170
08CH18N10T	expansion compensating pipe
Sv07CH25N13	vessel head cladding 1st layer
Sv04CH20N10G2B	vessel head cladding 2nd layer (anti-corrosive)
Sv07CH25N13	RPV cladding 1st layer
Sv04CH20N10G2B	RPV cladding 2nd layer (anti-corrosive)

Schematic representation of VVER 1000/213 reactor pressure vessel is shown in Fig. A.9 – A.12 in the Annex A hereto.

Methods and criteria used for selecting components within the scope of ageing management for reactor pressure vessels

In general, the scope for ageing management includes all components important to perform the functions relevant to safety or components, whose failure could affect the performance of the function of components relevant to safety in accordance with the SÚJB requirements and with the IAEA recommendations. Binding methods and criteria used for selecting of SSCs within the scope of ageing management are defined in the relevant documents (procedures) of the ČEZ company; for more details see Chapter 2 hereof.

Identification of ageing mechanisms for the individual materials and components of the reactor pressure vessels

The same methodology [26] for the ageing management review, which has been developed in accordance with the IAEA recommendations [6] and [7] and the SÚJB recommendations [5], is applicable to both reactor types (VVER 440/213 and VVER 1000/320).

In accordance with these documents, the Ageing Management Review (AMR) was carried out for the VVER 440/213 reactor pressure vessel in 2009 and again in 2014. The AMR for the VVER 1000/213 reactor vessel was developed in 2010 and revised in 2016.

The part of the AMR was evaluation of the the understanding of ageing, i.e. the ability to identify any existing and potential degradation mechanisms or ageing effects for individual reactor components.

Furthermore, ageing management programmes have been identified for every relevant degradation mechanism/ageing effect and were reviewed for their sufficiency for safe and economic operation including potential for long term operation. In the event that any deficiency has been identified, i.e. certain degradation mechanism/ageing effect was not adequately monitored, the appropriate remedial actions were proposed. These recommended remedial actions were implemented in order to ensure the required level of ageing management.

On the basis of the AMR carried out in 2009, the methodology for reactor life management has been developed – Component Specific Ageing Management Programme for Reactor [101], which was adapted to both reactor types later while respecting the relevant technical differences.

For the purposes of ageing management, the reactor was divided into its relevant components in this document:

- Pressure vessel, which comprises a cylindrical part including all nozzles and an elliptical bottom head
- Upper block, consisting of the vessel head itself and vessel head nozzles

- Flange, which consists of a part of the vessel head and of a part of the cylindrical vessel ensuring tightness of the main flange joint, of a floating flange and bolts
- Reactor internals
- Control rod drives

This division fully covers the components, for which the implementation of the overall AMP should be described according to the requirement referred to in Chapter 03.1.1. of the Technical Specification [1], i.e.:

- The steel vessel including base metal, cladding and welds;
- The vessel head and the lower dome including penetrations;
- Inlet and outlet nozzles.

In view of the scope of the Technical Specification [1], this chapter does not cover reactor internals and control rod drives.

The approach to reactor life management is described in the Component Specific AMP for Reactor [101]. The document includes the scope of monitored components including monitored functions, list of degradation mechanisms and ageing effects affecting the performance of function, which could affect the physical condition of these components, parameters for the monitoring of identified degradation mechanisms / ageing effects, and the method for summary assessment of the condition of reactor and its individual components. The following chapters include a detailed overview of the current status of ageing management of reactor pressure vessel.

For the AMR of reactor pressure vessel, the following specific information sources were used (while respecting the differences for individual vessel types):

- The operating experience documented in the Catalogue of Degradation Mechanisms specifically developed for the VVER 440 Dukovany NPP and the VVER 1000 Temelín NPP [39]
- Generic worldwide experience presented in IAEA SRS No. 82 [22] and previous document versions
- Generic operating experience from the USA: Generic Ageing Lessons Learned (GALL), NUREG-1801, rev.2, Office of Nuclear Reactor Regulation, 2010 [40] and previous document version
- Preventive Maintenance Database EPRI

In order to obtain information on the mode of operation, initial and actual condition of reactor components, the following documents were used:

- Design documentation
- Database with technological data
- Operating procedures
- Information obtained from the maintenance programs
- ISI Programme
- Chemistry Control Programme
- Specific Ageing Management Programmes:
  - AMP for Reactor Pressure Vessels in NPP [102]
  - AMP Low-cycle Fatigue [103]

- TLAA
- HEALTH Reports

The following chapters provide a detailed overview of the current status of ageing management of reactor pressure vessel for the selected NAR example: VVER 440/213 reactor pressure vessel of Dukovany NPP Unit 1, which was selected as the "NAR example" (see Chapter 05.1.1 [1]) because of earlier commissioning of that plant and due to that fact that it is currently at the stage of LTO.

5.1.1.2 Scope of ageing management for reactor vessel of the LVR-15 nuclear research reactor

The objective of ageing management is to ensure safe operation of the LVR-15 research reactor through a set of technical and organisational measures. As for reactor vessel, ensure the following functions:

- Maintaining integrity of the main boundary of reactor coolant
- Maintaining sufficient coolant level for core cooling in normal and abnormal operation
- Maintaining sufficient coolant level for core cooling during and after accident conditions, under which there was no failure of integrity of reactor coolant system

From the perspective of reactor vessel, integrity function is the function necessary for ensuring the above mentioned system functions. The reactor vessel is the selected equipment included in Safety Class 2 according to the Atomic Act.

The ageing assessment of the LVR-research reactor vessel takes place in accordance with the overall Ageing Management Programme for the LVR-15 Reactor, which was developed in 2008 and revised in 2011. The Programme has been developed with the use of the recommendations in IAEA-TECDOC-792 "Management of research reactor ageing" [31] and IAEA SSG-10 "Ageing Management for Research Reactors" [30]. As stated in Chapter 2, the Programme will be adjusted in the transition periods of validity of the Atomic Act so as to fully meet the requirements set out in new legislation.

<u>Description of the LVR-15 reactor vessel</u>

The <u>non-pressure</u> reactor vessel is made from 08CH18N10T stainless steel. The cylindrical baffle comprises several rings made up from sheet metal and longitudinally welded. The rings are welded among each other with circumferential welds, with the longitudinal welds being rotated one another always of 90°. The whole cylindrical baffle is then welded to the forged elliptical bottom with circumferential weld.

The vessel internals, i.e. horizontal channels, core support plate and core baffle (reactor core outer shell), are made from aluminium of a purity of 99 %.

The outer diameter of the vessel is 2300 mm, vessel wall thickness is 15 mm and vessel bottom thickness is 20 mm. The height of the vessel from the contact surface of the support plate is 5760 mm, vessel weight without water is 7900 kg, and water volume in the vessel is 22 m^3.

At the bottom of the vessel, there are two nozzles DN 300 x 16 for cooling water inlet, one nozzle DN 400 x 16 for cooling water outlet, one nozzle DN 100 x 16 for water overflow, one nozzle DN 100 x 16 for draining water from the vessel, one nozzle DN 20 x 16 for water sampling, one nozzle DN 10 for impulse line connection for vessel level measurement, one tube DN 125 serving as a chute for samples and one spare tube DN 70.

The support plate of the vessel is in circular ring shape, reinforced with ribs and welded to the vessel bottom. With this plate, the vessel is loosely laid on the cast iron support plate of the original VVR-S reactor, which is mounted in reactor cavity. The vessel was fabricated by Škoda Plzeň and was put into operation in the reconstruction of the original VVR-S reactor in 1989. The cast iron support plate and the concrete reactor cavity are original from 1957.

Identification of ageing mechanisms for the individual materials and components of the reactor vessel

In the framework of the development of the Ageing Management Programme for the LVR-15 reactor, an analysis of potential degradation mechanisms and ageing effects was carried out as well as life assessment for the selected components of the research reactor, which include reactor vessel and LVR-15 reactor internals, which is documented in the report DITI 304/268 [104]. Furthermore, the results of the periodic in-service inspections carried out under the approved In-service Inspection Programme were described and evaluated.

5.1.2 Ageing assessment of RPVs

5.1.2.1 Ageing assessment for reactor pressure vessels of the Dukovany and Temelín Nuclear Power Plants

As stated in Chapter 5.1.1.1., the ageing management of RPV in both plants is carried out under the Component Specific Ageing Management Programme for Reactors [101], while respecting design differences between the two reactor types. The outputs of other specific Ageing Management Programmes, i.e. AMP Low-cycle Fatigue [103] and AMP for Reactor Pressure Vessels [102] focused on the assessment of radiation damage to the RPV and other degradation mechanisms/ageing effects, are integrated in the Component Specific Ageing Management Programme for Reactor.

All programmes mentioned above are the controlled documents that are periodically updated in the light of internal and external feedback and the current state-of-the-art in this field.

For all the required RPV elements, for which the implementation of the overall Ageing Management Programme should be demonstrated under Chapter 05.1.1 of the Technical Specification [1], the following documents served as a basis for setting up the appropriate Ageing Management Programme:

<u>As-built drawing documentation:</u>

- Technical documentation
- Passports
- Specifications
- Quality Assurance Programmes

<u>In-service Inspections Programme (Supervision Programme)</u>

- No. 001 “Reactor”

<u>Conclusions of the Ageing Management Review 2009 for implementation of the Component Specific Ageing Management Program of RPVs (first “AMR”)</u>

- Life management matrix for 213-Č and its update under the TST_0033 [105]

Standards, codes and guides

- NTD of the Association of Mechanical Engineers Section IV − Residual life assessment of equipment and pipework for VVER nuclear power plants, NORMATIVE TECHNICAL DOCUMENTATION of the Association of Mechanical Engineers Brno, 2016 [105]
- VERLIFE: Guidelines for Integrity and Life time Assessment of Components and Piping in WWER NPPs during Operation, IAEA, 2008 [106]
- VERLIFE: Guidelines for Integrity and Life time Assessment of Components and Piping in WWER NPPs during Operation, IAEA, to be published in 2013 [107]
- Pravila ustrojstva i bezopasnoj ekspluatacii oborudovanija truboprovodov atomnych energetičeskich ustanovok (PNAEG-7-008-89) [108]
- Normy rasčota na pročnosť oborudovanija i trunoprovodov atomnych energetičeskich ustanovok (PNAEG-7-002-87) [109]
- Oborudovanije i truboprovody atomnych energetičeskich ustanovok. Svarka i naplavka. Osnovnye položenija. (PNAEG-7-009-89) [110]
- Regulations for testing welded joints and cladding of flanges and structures of nuclear power plants and nuclear research reactors and facilities (PK1514/72) Gosgortechnadzor, 1974 [111]

Assessment Report of AMR 2015 JCHO/LTOA/CP-YC1-14/39R0FR1

- Ageing Management Review for Dukovany NPP machinery for the needs of LTO, Assessment of the YC1 system – Reactor and reactor internals

Supporting documentation for strength, lifetime, seismic resistance:

– Seismic qualification:

SQ documentation:

- EQ-B1-1P-22/35 – ÚJV Řež ref.no. 10815, DITI 300/94, rep. 73-98.3ep, rev 1
- EQ-B1-2P-22/35 – ÚJV Řež ref.no. 10815, DITI 300/94, rep.73-98.3ep, rev 1
- EQ-B1-3P-22/35 – ÚJV Řež ref.no. 10815, DITI 300/94, rep.73-98.3ep, rev 1
- EQ-B1-4P-22/35 – ÚJV Řež ref.no. 10815, DITI 300/94, rep.73-98.3ep, rev 1

The following outputs of research programmes were used in the review of ageing management for the stage of review called understanding of ageing. The listed information sources (in their current forms) are then used in the ageing management of reactor as a source of generalised worldwide experience.

INCEFA

- For selecting the approach to environmental fatigue assessment in conformity with the developing state of the art in the world.

PROSAFE

- Development and application of the probabilistic safety assessment for reactor.

SOTERIA

- Development of the understanding of radiation and thermal ageing of RPV materials.

WPS - SÚJB

- Influence of hot overload on the integrity of reactor pressure vessel in accidents with pressurized thermal shock (PTS), SÚJB Project, implemented in ÚJV Řež, a.s. between 2006 and 2008

IAEA VERLIFE 2017

- The methodology contains outputs of the extensive research programme dealing with degradation of VVER reactor materials during operation and provides a knowledge base for degradation assessment.

Information from the EPRI, of which ČEZ, a.s., is a member, serves as a permanent source of information from research and development used for reactor operation and for assurance of LTO.

Information from the EPRI is used in a number of fields in reactor operation:

- For understanding the ageing as a source of information about the newly identified degradation mechanisms (e.g. swelling); in the field of methodology, for assessing sudden operation events (e.g. RPV corrosion due to reactor head overflowing).
- For identifying the most recent approaches in the field of in-service inspections of reactors and applying new procedures and technologies in in-service inspections (potential for the application of new methods – Phased array UT, verification of inspection procedures and methods for reactor internals – core baffle measurement, reactor core basket screw testing,…)

Internal and external operating experience

Internal and general external operating experience is used for the updating and verification of ageing management setting. Information is collected and analysed once a year.

Internal operating experience is evaluated on the basis of the occurrence of events with an impact on ageing management for RPV and reactor internals. Based on the existing information sources of the plant, the impact of identified operational events on reactor ageing and lifetime is assessed. The following information sources are used:

- Report for the meeting of the group of experts to evaluate the results of in-service inspections of Dukovany NPP Unit 1 (2, 3, 4) in the period of regular general overhaul in the current year and campaign of operation
- Report on the inspections carried out on Unit 1 (2, 3, 4) under the in-service inspection plan for the Dukovany NPP in the current year and campaign
- Reliability-based information system
- Reports of the Meeting of the Failure Commission of the Dukovany Nuclear Power Plant
- Records on compliance with the Limits and Conditions
- Maintenance information system Asset Suite AS 8

After the campaign of the unit in a given calendar year, the listed information sources are analysed as regards whether any event occurred during operation, which could potentially affect the operation and lifetime of reactor pressure vessels. In case of identification of such event, it is subsequently peer-reviewed and taken into account in ageing review of reactor pressure vessel, which is, based on the Component Specific Ageing Management Programme, developed once a year.

External operating experience, change in the state of knowledge, state of the latests knowledge of science and technology, and their relevance to ageing management are evaluated in the framework of the ageing management parameter "Technical Ageing". Knowledge is acquired during work on the projects implemented particularly in cooperation with the EPRI, NUGENIA and IAEA; the important source of information for ageing management is scientific conferences focusing to safe operation of nuclear power plants.

Regularly (once a year) held workshops on ageing management are another source of information about external operating experience. These workshops are attended by specialists from nuclear power plants in operation (ČEZ a. s., SE a.s.) and experts from supporting scientific-research organisations from the Czech Republic and the Slovak Republic (ÚJV Řež, a.s., VUJE).

If necessary for the issue of ageing for individual major components of the NPPs, "benchmark" workshops are also held, particularly for information exchange and comparison of individual specific assessment methods for equipment including differences in the philosophy of the approach to ageing management of reactor pressure vessels in both participating countries.

In addition, the technical ageing is evaluated on the basis of experience and operational events from other plants in the world. In recent years, information obtained from events in nuclear power plants was used for the evaluation of parameter:

- Shaeron Harris NPP
- Doel and Tihange NPPs
- Paks and Mochovce NPPs
- Beznau 1 NPP
- Loviisa NPP

In compliance with the methodology for internal and external operating experience feedback of NPPs, operating experience published by the WANO is used for feedback.

Identified degradation mechanisms and ageing effects including evaluation of their significance

On the basis of information sources, procedures and activities mentioned and described above, the following degradation mechanisms and ageing effects have been identified for RPV, which should be adequately managed to ensure long-term safe operation. These are:

- Radiation and thermal ageing
- Fatigue
- Stress corrosion cracking (SCC, IASCC)
- Corrosion (primary medium on the outer surface of the reactor)
- Loss of the preload on bolted joints caused by increased temperature
- Mechanical damage / Wear/ Abrasion

Radiation and thermal ageing

Radiation and thermal ageing is the significant degradation mechanism limiting reactor life. It is evaluated in the framework of monitoring of radiation and thermal ageing of RPV with the use of surveillance specimens. The criterion value, maximum temperature of brittleness T_k^a, was determined on the basis of the assessment of reactor vessel resistance to brittle fracture. The assessment was carried out in compliance with the requirements [NTD of the Association of Mechanical Engineers, VERLIFE] by means of PTS calculations according to the relevant methodologies.

Radiation and thermal ageing is assessed for the cylindrical part of pressure vessel in the area of reactor core for weld metals, heat affected zone, base material, cladding through the assessment of changes in temperature of brittleness Tk. Changes in Tk are assessed periodically, once a year, in the light of updated fluence values and outputs of the surveillance specimen programme, if any is available in a given year.

Fatigue (low-cycle)

Fatigue is the degradation mechanism potentially limiting reactor life. To ensure long-term life, the original design fatigue calculations were recalculated in order to specify and eliminate any excessive conservatism and the system for monitoring low-cycle fatigue of the Dukovany NPP (DIALIFE) was updated on the basis of such recalculations. Fatigue accumulation and limit values are determined in accordance with the NTD of the Association of Mechanical Engineers [105]. Fatigue development is assessed periodically, once a year. Increase in fatigue damage is assessed for all reactor components, for critical points on these components determined by calculation.

Stress corrosion cracking

Stress corrosion cracking is assessed on the basis of outputs of the In-service Inspection Programme. Critical points, detection methods and acceptance criteria are established in operating procedures for inspections, in individual instructions and guidance documents for specific inspection methods. When respecting the required water chemistry of the primary circuit (given by the relevant operating regulation) and other operational procedures, the development of stress corrosion cracking is unlikely. Stress corrosion cracking is assessed once a year for the components that are in contact with the corroding medium, specifically pressure vessel liner, vessel upper head liner and vessel head nozzle shrouds, relevant parts of the flange.

Corrosion

Corrosion is eliminated provided that the operating procedures are respected – contact of the primary medium with the outer surface of the reactor should be avoided. Any development of corrosion is identified under the In-service Inspection Programme (criteria are established in the operating procedures for individual inspections) and in the assessment of each event, which involved violation of operating regulations and contact of the primary medium with the outer surface. Corrosion development is unlikely. An expert opinion is drawn up on each contact. Corrosion development is assessed once a year.

Loss of the preload on bolted joints caused by increased temperature

This ageing effect concerns bolted joints on the reactor, in particular bolts of the reactor main flange. Its development is detected by dimensional inspections under the In-service Inspection Programme. It is a non-important ageing effect to ensuring long-term operation. The criterion acceptance value of the loss of preload is determined in the relevant instruction for inspection under the In-service Inspection Programme. The loss of the preload on bolted joints is assessed once a year.

Mechanical damage / Wear

Mechanical damage, wear or abrasion occurs during relative movement of metal parts. It may occur because of non-fulfilment of the operating procedures or during normal operation. The ageing effect is controlled by visual inspections under the In-service Inspection Programme; the relevant acceptance criteria for mechanical damage are established in the instructions for individual inspections. On the basis of experience from previous operation, a significant development of this ageing effect is not expected. It is assessed once a year.

The following degradation mechanisms affect specific RPV components:

Pressure vessel:

- IASCC, SCC
- Radiation and thermal ageing
- Fatigue
- Corrosion
- Wear

Upper block:

- SCC
- Fatigue
- Corrosion
- Wear

Flange:

- SCC
- Fatigue
- Corrosion
- Wear

Individual degradation mechanisms acceptance criteria establishment process

Radiation and thermal ageing

The limit value of critical temperature of brittleness T_k^a was determined under the "Assessment of pressure vessel resistance to pressurized thermal shock" carried out between 1998 and 2006. The assessment of critical regimes was updated between 2007 and 2008. The normal values of temperature of brittleness T_k are derived from the value of T_k^a and represent a degree of material embrittlement, when the operation is possible without taking remedial actions. The maximum normal value of T_k (acceptance criterion) is determined in the framework of activities under the Component Specific Ageing Management Programme for Reactor and is lower than the limit value of T_k^a.

On the basis of outputs of the Specific Ageing Management Programme for RPVs [101], the reassessment of the resistance of reactor pressure vessel to pressurized thermal shock was initiated in 2015. This Programme was initiated due to change in the state of knowledge over the past ten to fifteen years – development of new assessment methods for PTS, updating of normative technical documentation, development of new calculation codes for thermohydraulic analyses and change in plant configuration including change in the relevant emergency operating procedures.

The output of this Programme in 2020 will be the specified value of T_k^a, determined in compliance with the current "state of the art" of the issue, taking account of the current plant configuration.

Fatigue (low-cycle)

The criterion value of fatigue accumulation is determined in the Normative Technical Documentation of the Association of Mechanical Engineers of the Czech Republic [105]. It is a living, continuously updated, document. Development in the issue and requirements of the industry –

operation are taken into account in the updates. The criterion acceptance value is taken from the NTD of the Association of Mechanical Engineers [105] and is indicated in the Component Specific AMP for Reactor [101].

Abrasion, Corrosion, Stress corrosion cracking including IASCC

The criterion acceptance values are determined in the relevant instructions and guidance documents for inspections in the In-service Inspection Programme for the detection of the specified degradation mechanisms / ageing effects. The In-service Inspection Programme has been developed on the basis of the accompanying technical documentation of the manufacturer, on the basis of manufacturer's and designer's recommendations and on the basis of the standards valid at the time of manufacturing. The In-service Inspection Programme is periodically modified including the scope of inspections, used methods and appropriate criteria. The modifications are made on the basis of operating experience and development of world knowledge of that issue. The In-service Inspection Programme is approved by the SÚJB.

5.1.2.2 Ageing assessment of non-pressure reactor vessel of the LVR-15 nuclear research reactor

In addition to the analysis described in Chapter 5.1.1.2, other sources served as a basis for setting up the Ageing Management Programme, e.g. drawing documentation, vessel passport, Initial Safety Analysis Report; Final Safety Analysis Report, operating conditions, neutron fluence calculations for the selected areas of reactor vessel.

Internal and external operating experience

For eventual updating of the ageing management programe, the results of inspections, testing and maintenance are used. In addition, the results of research projects, special inspections and checks and information from the operation of similar facilities are used.

Identified degradation mechanisms and ageing effects including identification of their significance

In designing and constructing the LVR-15 research reactor, all materials were selected particularly with regard to ensuring high corrosion resistance in the environment of flowing treated water of the primary circuit under normal pressure and under different radiation conditions.

Titanium and aluminium alloy stabilised austenitic stainless steel 08CH18N10T is the main material used for reactor vessel and internals.

As mentioned above, possible degradation mechanisms were assessed in the report "Life Assessment of Selected Components of LVR-15 Research Reactor" [104], with their identification based on expert knowledge of the team from ÚJV Řež a.s., which conducts similar assessments for nuclear power plants.

At maximum temperature of 70°C, material damage caused by thermal ageing and creeping including swelling can be practically eliminated. Hydrogen and helium embrittlement can be included among the processes that require high temperatures. This degradation mechanism, which primarily depends on diffusion, also does not apply under the conditions of research reactor.

Possible degradation mechanisms are: corrosion (erosion), including chemical processes, fatigue, including vibrations, and radiation damage.

Radiation damage

Vessel and grates are made from austenitic stainless steel 08Ch18N10T and 17246.4. Welds are made with the filler material Sv04Ch19N11M3 of the same type. The maximum neutron flux at the vessel wall is 1.5×10^{10} n/cm2/s.

The values of fluence in standard operation of the LVR-15 reactor were calculated until the end of 2018, i.e. after 29 years of operation and until 2030, which is the expected extension time of operation, and four most exposed points on reactor vessel and internals were selected depending on operating time. The values of fluence range mainly below the threshold value for possible radiation damage to austenitic steels (including the year 2030).

The values of fluence show that the least radiation load is applied to the weld of outlet pipe and vessel wall. The expected, in an order of magnitude greater, load was detected for certain internals near the reactor core. The maximum fluence value until 2030 was calculated at the place of beginning of the outlet line at the core support plate. This value approximates the lower limit of potential radiation damage to austenitic materials, which is based on the fluence threshold values indicated for radiation damage to 08Ch18N10T steel in an order of magnitude of $10E^{20}$-$10E^{22}$ n/cm2, see "Prediction of Mechanical Properties of Irradiated Austenitic Stainless Steels", ENES, Moscow 2007 [113].

Corrosion

For inner surface of the vessel itself, no excessive damage has been demonstrated, which is related to surface or point corrosion.

For internals, equipment made from aluminium is potentially problematic and is susceptible to point corrosion. In terms of corrosion damage, horizontal channels are the most critical parts, for which loss of integrity can lead to coolant leakage from reactor vessel. There are points suspected of point corrosion. This type of damage was repeatedly detected on the outer surface of channels in different locations.

Flange seal of the horizontal channels is also critical in terms of corrosion. Therefore, surveillance model flange was designed, which could be used for condition monitoring of flanges in the vessel. This flange can be generally regarded as the most critical of the assessed components of reactor vessel and internals.

Fatigue life

Static strength and fatigue life were calculated for hydrostatic pressure and temperature field load under the former assessments (Check Calculation DRS-867, 1995). The conditions of static strength and fatigue life are satisfactory.

Process for the establishment of acceptance criteria for individual degradation mechanisms

Determination of acceptance criteria is based on the results of safety analyzes and the resulting operating limits and conditions; for the reactor vessel the limit value is the total neutron fluence. In the analysis [104] the possibility of reaching the limit parameters of the structural components of the key components was assessed.

5.1.3 Monitoring, testing, sampling and inspection activities for RPVs

5.1.3.1 Monitoring, testing, sampling and inspection activities for reactor pressure vessels of the Dukovany and Temelín Nuclear Power Plants

Low-cycle fatigue monitoring programme

The first assessment of fatigue damage was carried out already at the design stage and forms a part of the accompanying documentation (strength and fatigue calculations). Critical (the most stressed) points were selected in the framework of the calculations. The initial selection was specified during operation, most recently in the comprehensive fatigue reassessment of pressure vessel between 2012 and 2014.

On the basis of operational regimes history (determined by in-process measurements), fatigue accumulation for the relevant campaign and subsequently overall fatigue accumulation are determined for each critical point for the overall period of operation, i.e. the period of assessment is once a year. The acceptance criteria are established in conformity with the requirements of the NTD of the Association of Mechanical Engineers, Section IV [105] in the Technical Standard for Ageing Management [101]. Condition assessment of the fatigue damage accumulation includes the prediction of development for the planned period of operation. The current fatigue accumulation is determined and predicted for each point being assessed with the use of the DIALIFE software application.

The critical points (with the highest fatigue accumulation) for individual reactor pressure vessel components are:

- Upper block: TM nozzle flange, cladding in nozzle flange on the sealing surface
- Pressure vessel: flow distribution node
- Flange: bottom plane of floating flange

Programme for monitoring operating parameters important to safety

Any unauthorised changes in operating parameters important to safety are periodically (once a year) evaluated with respect to their impact on ageing. The acceptance criteria are defined in the operating regulations “Limits and Conditions of Safe Operation”.

Maintenance

The activities carried out, among others, as part of preventive maintenance serve to monitor reactor physical condition and to evaluate the effects of operating conditions and degradation mechanisms on reactor. The original period and criteria for individual in-service inspections are given by the requirements for remedy and monitoring of the effects of degradation mechanisms specified by the manufacturer in the Technical Specifications and in the Individual Quality Assurance Programme for reactor. During operation, they were updated on the basis of operating experience and worldwide experience.

The assessment of preventive maintenance is based on the operating procedures for repairs and the maintenance records from the ISE PassPort system.

The current preventive maintenance plan with regard to ageing management (including periodicity of actions) is defined in the Component Specific Ageing Management Programme for Reactor [101] and is reflected in the control application ISE PassPort. The periodicity of individual activities ranges from the interval of once a year to every eight years.

The outputs of maintenance are periodically (once a year) evaluated in order to assess development of potential degradation mechanisms that have not been identified by other effective Ageing Management Programmes and other processes. The acceptance criteria are set out in the relevant operating procedures. Both preventive maintenance and random maintenance activities, if any, are reviewed.

In maintenance assessment, activities and findings of each year are compared in order to reveal any potential trend of progressive degradation. No trend has yet been observed.

In-service Inspection Programme

The scope of in-service inspections includes the selected inspection points of all reactor components.

The In-service Inspection Programme for reactor is based on the original Soviet regulations for NPPs, Technical specifications and on the Individual Quality Assurance Programme (IQAP) for reactor established by the manufacturer.

The scope of the inspections prescribed in the In-service Inspection Programme is regularly updated in accordance with the needs, requirements and current knowledge of the component in question and its associated degradation mechanisms, specifically on the basis of the findings of reactor condition assessment, operating experience gained in the Czech Republic and in the world, SÚJB recommendations and the requirements stemming from the documents VERLIFE and NTD of the Association of Mechanical Engineers. Furthermore, with regard to the development of non-destructive measuring technology, new technologies are adopted and new devices are used, enabling the improvement of the outputs obtained from in-service inspections.

The in-service inspections are carried out to the extent and in the period to ensure condition monitoring or detection of the effects of degradation mechanisms such as:

- Stress corrosion cracking
- Low-cycle fatigue
- Radiation embrittlement
- Mechanical wear
- Corrosion

The inspections are carried out with the use of ultrasonic pulse-echo testing method and time-of-flight diffraction (TOFD) ultrasonic testing method (welds and base material of RPV), eddy-current testing method and indirect visual testing method with the use of high-resolution cameras. Welds and base material of reactor pressure vessel in the Dukovany NPP are tested every four years, alternately from the outside and the inside. Ultrasonic testing and eddy-current testing are carried out in a qualified way.

Welds between the extension piece and the main circulation piping are inspected from the outside by ultrasonic, visual and penetration testing methods at an interval of 96 months, and from the inside by visual, ultrasonic (pulse-echo and conditionally TOFD method) and eddy-current testing methods also at an interval of 96 months.

If required, the inspection programme is expanded to cover other inspection points and, if appropriate, the scope of inspection is adjusted or a new method is added for the existing inspection points (e.g. inspections added for VVER 440 in 2015 for upper block - sealing surface Ø 172 mm / 180 mm of the middle flange of transport corridor, NFM nozzle - visual and penetration testing methods, for floating flange – contact surface - visual and penetration testing methods).

The inspection of reactor pressure vessel is performed by the “MKS” automated system (testing from the inside) and the SK-187 system (testing from the outside). The “Traktor-HB” manipulator is used for the inspection of weld on pressure vessel head.

Other inspections associated with refuelling are carried out at an interval of 12 months, 36 or 48 months.

The inspections carried out at an interval of 12 months are mainly associated with the loss of reactor integrity, e.g. main parting line inspections. Claddings on sealing surfaces and grooves of reactor main flange are tested with the use of visual and penetration testing methods; dimensional inspection is carried out for grooves by 200 mm.

Bolts M140x6 are inspected at threads and also in the cylindrical part. Bolts are inspected every 12 months with the use of the dimensional inspection and visual testing/inspection and are conditionally subject to penetration testing. Inspections by ultrasonic and eddy-current methods are performed on the cylindrical part and thread of the screws at an interval of 96 months.

The results of in-service inspections are always recorded in the report containing all relevant information for the unique identification of the area under inspection, used methods, procedure, evaluation and inspection result. The reports of all inspections are archived both in paper and electronic form throughout the operation. The acceptance criteria are defined in accordance with the standards, by the manufacturer and on the basis of the requirements set out in the NTD of the Association of Mechanical Engineers.

Generally speaking, the outputs of the In-service Inspection Programme are periodically (once a year) analysed considering all information available (not only for the Dukovany and Temelín NPPs but also all information published in the world). The results of in-service inspections recorded in reports are compared with the results obtained in previous years. The activities carried out as part of the In-service Inspection Programme serve to monitor reactor condition and to evaluate the effects of operating conditions and degradation mechanisms on reactor. The risks of development of the identified degradation mechanisms as well as the potential for development of the degradations not yet expected are identified in the assessment.

Surveillance Specimen Programme

Due to radiation effects, pressure vessel material embrittlement occurs in the area of reactor core. Therefore, it is necessary to monitor material changes in RPV during operation. The design of each of the reactor pressure vessels (RPV) of VVER-440/V-213Č type already ensured the surveillance specimen programme for materials of the pressure vessel. This programme was proposed by major design organisation, i.e. OKB Gidropress, Podolsk, already at the turn of 60s and 70s on the basis of the knowledge and possibilities of that time in the former Soviet Union and with regard to reactor design, i.e. mutual configuration of pressure vessel and its internals.

The original, so-called “Standard Surveillance Specimen Programme”, implemented in the Dukovany NPP in compliance with the accompanying technical documentation, did not entirely satisfy the current requirements for the content, purpose and required results, mainly in terms of their applicability to the assessment of the residual life of reactor pressure vessel, in particular in the following aspects:

- High coefficient of acceleration of the neutron flux on surveillance specimens with respect to inner wall of reactor vessel, (approximately 10);

- Inaccuracies in the determination of the actual flux on individual surveillance specimens relative to their orientation to the centre of reactor core;
- Inability to determine the actual exposure temperatures;
- Inability to use the results of fracture toughness test;
- Absence of cladding material;
- Period of monitoring the effect of operation of pressure vessels on changes in their material properties limited to five years.

In order to remedy such deficiencies and obtain reliable data, a new surveillance programme was proposed (ÚJV Řež, a.s. and Škoda JS a.s.), taking into account all the above aspects in its concept, the so-called “Supplementary Surveillance Specimen Programme (SSSP)” for reactor pressure vessels of the Dukovany NPP.

The purpose of the SSSP is to ensure the possibility of residual life assessment for reactor pressure vessels for their design period of operation in accordance with the “Guidelines and recommendations regarding ageing assessment of NPP VVER reactor pressure vessel and internals during NPP operation” [112].

The evaluated test and measurement results serve as a basis for identifying the following supporting documents needed for the assessment of RPV residual life:

- Continuous time dependence of changes in yield strength, tensile strength, shift in critical temperature of brittleness from notch toughness tests and transition temperature from fracture toughness tests, with knowledge of the actual exposure temperature
- Continuous time dependence of the surveillance specimens neutron fluence (integrally always after several campaigns)

The SSSP was not proposed in terms of the requirements of operation beyond the original design life. The Extended Surveillance Programme for the materials of reactor pressure vessels of the Dukovany NPP was prepared to meets, with certain reserves, the requirements for providing the necessary data for life extension of all reactor pressure vessels of the Dukovany NPP.

The programme is primarily used for the following purposes:

- Qualification of representative weld metals
- Monitoring of neutron flux at the RPV wall after increase in power level
- Monitoring of radiation embrittlement of RPV base materials for continued life
- Monitoring of radiation embrittlement of RPV weld metals for continued life
- Monitoring of radiation embrittlement of heat affected zone of RPV welds for continued life
- Monitoring of radiation damage to austenitic cladding of RPV for continued life
- Weld repairs

Surveillance specimens are stored in six pairs of chains with the container containing surveillance specimens; each container contains 6 tensile test specimens, 12 inserts for Charpy test specimens with notch or Charpy test specimens with pre-cycled crack for static fracture toughness testing. The programme also covers base material, welds including heat affected zone, first and second layer of cladding, and reference steel JRQ. Furthermore, each container contains fluence detection monitors and temperature monitors. In addition, the Extended Surveillance Programme includes fluence monitoring outside the reactor vessel. The containers are placed in the area of reactor core and outside it in the area of upper ring to be able to monitor the influence of neutron and thermal embrittlement. Removing containers and evaluating specimens take place in accordance with the pre-approved and, where necessary, updated time schedule to be able to assess the effects of neutron and thermal ageing for the whole planned operation of pressure vessel.

The outputs of the surveillance programme are evaluated once a year. The results of neutron dosimetry and mechanical properties of the materials after irradiation are periodically updated and stored in the database of the Surveillance Programme. The current actual material brittleness temperature T_k and its future prediction according to the method defined in the standard [105] are the final output.

In the Component Specific Ageing Management Programme for Reactor, the acceptance criterion derived from the maximum allowable brittleness temperature T_k^a is defined. Depending on the current and predicted value of T_k and values of T_k^a, operational safety of the reactor pressure vessel is demonstrated with regard to thermal and neutron ageing and, where necessary, the appropriate remedial actions are defined.

5.1.3.2 Monitoring, testing, sampling and inspection activities for reactor vessel of the LVR-15 nuclear research reactor

Condition monitoring of the components is based on the approved in-service inspection programme, where the individual inspections are defined including interval of such inspections. The document is regularly updated in accordance with the needs, requirements and current knowledge of the component in question, or in line with the development of the methods concerned. The inspection activities include:

- In-service non-destructive testing of vessel and pipework materials
- In-service non-destructive testing of welds
- Visual surface inspections
- Tightness of the flange of horizontal channel

5.1.4 Preventive and remedial actions for RPVs

5.1.4.1 Preventive and remedial actions for reactor pressure vessels of the Dukovany and Temelín Nuclear Power Plants

In the Component Specific Ageing Management Programme for Reactor, defined by the relevant technical standard [101], parameters for ageing management are established including acceptance criteria and the so-called “normal values”. Exceeding of the normal values does not prevent the continued operation, but it leads to the requirement for the definition and implementation of remedial actions to address this situation and to ensure that the value of the parameters for life management in continued operation is again in the range of normal values.

The individual recommendations for remedial actions are defined in the assessment of individual parameters and their final definitions are summarised in the document “Periodic Life Assessment of Reactor” developed on a yearly basis.

Implementation of remedial actions is then reviewed in the next assessment of the parameters for ageing management.

In the past, subsequent remedial actions have been defined and implemented in order to ensure the required life of reactor pressure vessel:

- Optimisation of fuel loading pattern in order to obtain the so-called “low-leakage zone” with minimum neutron load of the reactor pressure vessel wall (reduction of up to 50% of fluence was achieved). This optimisation was carried out following campaign 12. After increase in power level to 500 MW, reduction in fluence is approximately 40% against the initial design value.
- Periodic updating of the requirements for pressure and temperature during reactor startup/shutdown in order to minimise the risk of cold overpressure. The update was and is carried out on the basis of the outputs of the Ageing Management Programme, in particular on the basis of material properties of the pressure vessel – the current and predicted brittleness temperatures.
- Definition of the appropriate operator actions (modifications of Emergency Operating Procedures) to ensure the resistance to brittle fracture during PTS event. Potential operator actions were analysed and subsequently defined on the basis of assessment of individual PTS scenarios as part of the PTS analyses for the determination of maximum allowable brittleness temperature. The first assessment of the resistance of RPV to brittle fracture was implemented between 1996 and 2004, followed by the review of critical scenarios for the parameters of increased power output between 2008 and 2009. Reassessment of the resistance to PTS events has now been - on the basis of the outputs of the Component Specific Ageing Management Programme for Reactor - initiated because of new valid methodologies and with regard to the current plant configuration. The main outputs are expected by the end of 2018.
- Performance of maintenance in accordance with the relevant procedures in order to avoid contamination of the outer surface of reactor pressure vessel by primary medium and other chemical compounds containing halides, which can result in the development of corrosion.
    - Development of corrosion from the outside is periodically monitored under the In-service Inspection Programme
    - Any potential event identified under the Component Specific Ageing Management Programme in the evaluation of the parameter “Preventive Maintenance”
- Reduction in reactor heat-up and cool-down rate during startup/shutdown from/to parameters in order to reduce the fatigue stress of individual components
- Decrease in overpressure during pressure strength test in order to reduce the fatigue stress of individual components
- Periodic specification of the neutron irradiation of RPV materials. One of the above mentioned actions was the application of the principle of the so-called “low-leakage zone”. Zone configuration is periodically (once a year) planned on the basis of the calculations with the use of specialised calculation codes based on the following inputs:
    - Source terms of individual fuel assemblies
    - Neutron monitor measurements in surveillance specimen containers
    - Neutron monitor measurements outside the pressure vessel

- Moderation of neutron flux by passage through the vessel wall determined by calculation is correlated with the results from irradiation experiments and measurements on the LR-0 research reactor

5.1.4.2 Preventive and remedial actions for reactor vessel of the LVR-15 nuclear research reactor

In 2009, the lifetime assessment of selected components of the LVR-15 research reactor was conducted, which is documented in the report DITI 304/268 [104]. The assessment was carried out for the LVR-15 reactor vessel and the components of reactor internals in terms of ageing effects. Possible degradation mechanisms have been identified and their influence on these components was evaluated; furthermore, the results of periodic in-service inspections were described and evaluated as part of the inspections under the approved In-service Inspection Programme.

The report includes the actions proposed in the area of inspections and structural-technological modifications, which should ensure continued safe operation and long-term life of the LVR-15 reactor, and which are continuously updated and implemented:

- Adding periodic visual inspection (once a year) and ultrasonic NDT method (every five years) for corrosion attack on the surface of horizontal channel no. 1 in the In-service Inspection Programme.
- In 2008, specific “surveillance programme” established for flange seal of the horizontal channels, as the most critical part of the assessed components of that research reactor
- Making prints in the surface of wet container

5.2 Operator’s experience of the implementation of AMP for RPVs

5.2.1 Operator’s experience of the implementation of AMP for RPVs of the Dukovany and Temelín Nuclear Power Plants

As mentioned above, reactor (including reactor pressure vessel) ageing is assessed once a year through the evaluation of the individual ageing management parameters. Normal and limit values are defined for each parameter. Exceeding the normal parameter initiates actions that should ensure that the critical value of parameter is not exceeded.

Generally speaking, the development of degradation mechanisms and ageing effects meets the expectations. Changes in the assessment methodologies of the development of degradations and in the actual Ageing Management Programmes or any other processes involved were particularly made due to the development of the state of knowledge, Czech and, where appropriate, worldwide technical legislation (methodologies, guidelines, standards).

The following are examples where the actual Ageing Management Programmes had to be modificated on the basis of external or internal experience:

Maintenance

In the framework of bolts M140 and critical locations on the reactor main of main parting line fatigue damage monitoring, development of fatigue damage has been identified, which could lead to achieving and exceeding the limit values of fatigue accumulation during the LTO phase. As a result, the following actions were implemented:

- Performance of more precise fatigue analysis
- Change in the operating procedure for tightening of M140 bolts main parting line studs to ensure more uniform fatigue load of individual bolts.

Surveillance Specimen Programme

The gradual development of the Surveillance Specimen Programme from the Standard Surveillance Specimen Programme to the Supplementary and subsequently Extended Surveillance Specimen Programme was described in Chapter 5.1.3.1.

The transition from the standard programme to the supplementary programme was due to the development of state-of-art of the given issue, for example stricter requirements for lead factor (high coefficients of irradiation acceleration became non-conservative, i.e. unacceptable), requirements for more specific identification of the fluence and irradiation temperature obtained, missing materials and impossibility of using the static fracture toughness testing.

The transition from the supplementary programme to the extended surveillance programme was due to the requirements given by operation beyond the original design lifetime.

Under the Extended Surveillance Specimen Programme, the procedures for assessing material brittleness transition temperature were updated to meet the requirements of the updated version of the VERLIFE method under development considering the higher coefficients of safety.

In-service Inspection Programme

The In-service Inspection Programme is continuously updated in the light of available knowledge of degradation of the individual components of reactor pressure vessel. Inspections of floating flange in the area of contact surface were supplemented with visual and penetration testing; conditional eddy-current testing was put in place in the area of contact surface of the vessel head; furthermore, visual and penetration testing of the sealing surfaces of middle flanges were added for TM-NFM nozzles.

5.2.2 LVR-15 nuclear research reactor operator’s experience of the implementation of the AMP for reactor vessel

In operator´s opinion the reactor vessel care program is properly set up, but the overall ageing program will be updated to meet all newly established requirements of the new atomic legislation within the period of transitional provisions validity.

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

5.3.1 Regulator’s assessment and conclusions on Ageing Management Programme for Dukovany and Temelín RPVs

The SÚJB reviewed information concerning the Ageing Management Programme for Reactor that was provided for the purposes of this report by the operator of the Dukovany NPP and the Temelín NPP, together with information obtained from its assessment and inspection activities.

The activities focused on the monitoring of current condition and lifetime assessment of reactor pressure vessel were carried out from the beginning of operation and were expanded on the basis of the current state of knowledge, and external and internal feedback in this area. The current Component Specific Ageing Management Programme for Reactor covers all significant and expected degradation mechanisms and is set in accordance with the international best practices. The results of the periodic life assessment of reactor are presented in the Final Safety Analysis Report updated once a year and reviewed by the SÚJB. In the framework of inspection activity, particular attention is paid to the current results of in-service inspections and maintenance practices, any modifications and repairs during refuelling outages on individual units.

Last but not least, the Ageing Management Programme for Reactor was thoroughly reviewed during the licensing process for license to operate individual units of the Dukovany NPP after 30 years of operation (i.e. for "LTO"). No current outstanding areas for improvement in the method of ageing assessment of reactor pressure vessel have been identified within this process.

The Ageing Management Programme for Reactor meets the requirements set out in applicable legislation and other documents within the scope of national legislative and regulatory framework (see Chapter 2.1).

For the reasons set out above, the SÚJB considers the Component Specific Ageing Management Programme for Reactors of the Dukovany and Temelín NPPs to be properly set and sufficiently effective.

5.3.2 Regulator's assessment and conclusions on Ageing Management Programme for LVR-15 reactor vessel

The reactor vessel is a part of the overall ageing management programme for reactor LVR-15.The condition of the vessel and other important equipment was analyzed in the report [104]. The AMP was revised based on the results of this analysis. Also some remedial measures were determined (e.g. the surveillance sample insertion). The prediction of radiation damage was performed and results doesn´t reach the limiting threshold. Based on available data, SÚJB evaluation and inspection activities, it can be said, that the reactor vessel care program is suitably set. However, the whole process of ageing management will be reassessed by the SÚJB after the expiration of validity of new Atomict act transitional provisions.

6. Calandria/pressure tubes (CANDU)

No CANDU type reactor is in operation in the Czech Republic.

7. Concrete containment structures

A total of six power reactors equipped with reinforced concrete containment is in operation in the Czech Republic. Four VVER 440/213 reactors are located in the Dukovany Nuclear Power Plant (Dukovany NPP) and are equipped with reinforced concrete containment with a passive vacuum-bubbler system. Two VVER 1000/320 reactors are located in the Temelín Nuclear Power Plant (Temelín NPP) and are equipped with pre-stressed reinforced concrete full pressure containment.

The LVR-15 nuclear research reactor is placed in a concrete reactor shaft situated inside the reactor building. The reactor building is designed as a simple steel hall structure and the reactor is not equipped with a containment system. The combination of a steel supporting structure and concrete reactor shaft fulfils the protective function of reactor and the function of biological shielding at once. The information on civil structures provided for the research reactor LVR-15 are beyond the requirements specified in [1].

7.1 Description of Ageing Management Programme for reinforced concrete containments

7.1.1 Scope of Ageing Management Programme for reinforced concrete containments

7.1.1.1 Scope of Ageing Management Programme for reinforced concrete containments of the Dukovany and Temelín Nuclear Power Plants

<u>Reinforced Concrete Containment of Dukovany NPP</u>

The Dukovany Nuclear Power Plant is structurally split into two identical twin units. Each twin unit includes the ventilation stack. Each twin unit is further divided into two self-functioning reactor units.

From the point of view of their function, the structures of twin unit reactor buildings are divided into containment area and the non-hermetic part. The containment boundary is defined by the position of the steel hermetic liner which provides integral seal tightness of the hermetic zone in case of the design basis accident associated with the loss of the primary circuit integrity.

The containment is used to locate the radioactive substances in the hermetically closed area where all important nuclear technological equipment of the generating process is installed, namely reactor, primary circuit, main circulation pump, steam generators and a number of other equipment. The main parts of the hermetic zone include the steam generator boxes, reactor cavity, ventilation centre, connecting corridor and vacuum-bubbler condenser. The hermetic zone boundary is also formed by the containment penetrations, reactor protective shielding cap, equipment hatches, hermetic doors and the walls of the refuelling pool and the spent fuel pool.

The Dukovany NPP reactor building diagram is illustrated in Figure A.13 of Annex A hereof.

Reinforced Concrete Containment of Temelín NPP

The Temelín NPP is designed with two identical reactor units. Each reactor unit works independently. The reactor building protects the reactor and the primary circuit equipment against external hazards (climatic and seismic influences, terrorism, etc.) and forms the last barrier against the release of radioactive substances into the environment in case of accident. Each main generating unit includes the ventilation stack.

The containment is used to locate the radioactive substances in the hermetically closed area where all important nuclear technological equipment of the generating process is installed, namely reactor, primary circuit, main circulation pump, steam generator and a number of other equipment. The hermetic zone of reactor unit in the Temelín NPP is formed by a pre-stressed reinforced concrete structure. The containment is pre-stressed by a system of unbonded pre-stressed cables placed in cable channels. The cylindrical part of the containment is pre-stressed using 96 cables and the dome is pre-stressed with 36 cables. Both cable systems of the cylinder and the dome are anchored in the cornice. Pre-stressing cables are braided of the high carbon steel wires with low relaxation; the diameter of the wire is 5.0 mm and the number of these wires in each cable is approximately 450. The cables of the cylindrical part are anchored in the top edge of the cylinder in the bearing ring and are guided in the helix shape to the bottom edge of the cylinder where they are bent and return back to the anchorage on the top edge of the cylinder. This system of cable guiding ensures pre-stressing of the structure both in the longitudinal and radial directions. The dome is pre-stressed with similar cables in two directions perpendicular to each other; the cables are guided in the dome area shape. These cables are anchored along the circumference of the cylindrical part in the bearing ring of the containment.

The Temelín NPP reactor building diagram is illustrated in Figure A.14 of Annex A hereof.

Containments of both NPPs fulfil especially three basic safety functions:

- Prevention of radioactive substance spreading outside the hermetic zone, representing the last barrier in the defence in depth,
- Protection of equipment the failure of which may result in the leak of radioactive substances; protection against external influences
- shielding

The containment structure fulfils the safety functions if the following conditions are met:

- The structural materials of the containment (i.e. concrete, concrete reinforcement, pre-stressing reinforcement, elements of the anchorage systems and the steel lining) are free of any defects endangering their functions.
- The sufficient level of the structure pre-stress is achieved (applicable to the Temelín NPP with the pre-stressed containment only. The containment of Dukovany NPP is without the pre-stressing system).

With respect to the essential importance of the containment in the system ensuring the safety of the nuclear power plant operation, it is necessary to ensure fulfilment of all required functions of the containment throughout the period of the NPP operation.

The containment structure is subject to the degradation effects. Hence, the condition of the structure is continuously monitored and its ability to meet the design functions is continuously evaluated.

In case of the Temelín NPP, a pre-stressed reinforced concrete structure is also considered and, therefore, its sufficient pre-stress is a condition for ensuring the strength function of the structure. Hence, the attention is paid to changes in the pre-stressing force (pre-stress losses). The In-service Inspection Activities Programme was developed already in the design phase and the structure was fitted with the sensors of a several measuring systems which enable monitoring of changes in deformation, stress and the level of pre-stress in the period of time.

The internals of the containment fulfil the static function and ensure:

- Ability to transfer its load to the installed equipment (such as the reactor pressure vessel, pools with double lining, steam generators and all primary circuit equipment) in the position defined by the design and under the extreme external conditions;
- Ability to withstand the static and dynamic loads caused by the operation of technological equipment inside the containment,
- Ability to protect this equipment against the effects of external loads without permanent deformations of the bearing structure itself and the supported equipment.

The structure of the containment internals fulfils the static function if the following conditions are met:

- The sufficient levels of stiffness and positional stability of structures below the containment bottom are achieved.
- Structural materials of the containment internals (i.e. concrete, concrete reinforcement and steel lining) are free of any material defects.

With respect to the essential importance of the containment internals in the system ensuring the safety of the nuclear power plant operation as a support for the primary circuit technology, it is necessary to ensure fulfilment of all required functions of the containment throughout the period of the NPP operation. Whereas the structure of containment internals is subject to the degradation effects, its condition is monitored and its ability to meet the design functions is evaluated on an on-going basis. It is the reinforced concrete structure the carrying capacity of which depends on the amount and condition of the concrete reinforcement.

Whereas the structures are subject to various degradation mechanisms during their life time, such as corrosion, aggressivity of the ambient environment, etc., ageing of civil structures on both power plants is managed in an appropriate manner as described below.

General rules, principles and methodology of ageing management processes are provided in Chapter 2 hereof.

Methods and criteria used for selecting components within the scope of the ageing management of the Dukovany NPP and Temelín NPP

General rules, principles and methodology of component selection are provided in Chapter 2 hereof. Component division and selection was carried out upon the engineering judgement with respect to the worldwide practice based on US NRC GALL, IGALL Safety Report, EPRI, IAEA, ACI. The following selection criteria were applied:

- Civil structures the parts of which are classified in SC 2 or SC 3 pursuant to the Atomic Act
- Economically important civil structures.
- Civil structures important from the viewpoint of protection of nuclear safety equipment
- Civil structures classified as per the worldwide practice

Scope of the AMP for the Dukovany NPP:

Generally, ageing of civil structures of the Dukovany NPP is managed by means of the Ageing Management Programmes (AMPs) described hereafter. These programmes help to monitor adverse effects of degradation mechanisms on the physical condition of civil structures and to predict the trend of future development. Thanks to that, the effective preventive or corrective actions can be taken to eliminate adverse effects of civil structures ageing and to ensure reliable fulfilment of their design and safety functions.

The Ageing Management Programmes and their development and general rules, principles and methods of the component selections are described in detail in Chapter 2 hereof.

For the purposes of below listed Ageing Management Programmes, the civil structures were divided into the structures/components of which they are made (e.g. reinforced concrete structures, steel structures, etc.) and the degradation mechanisms which affect individual structures/components throughout their life time were allocated to individual structures/components.

As a part of individual AMPs implemented in the Dukovany NPP, the degradation mechanisms affecting the structures are provided including their expected adverse impact. These degradation mechanisms are identified in order to evaluate and mitigate the effects of degradation of civil structures.

The periodic visual inspections are carried out to identify the degradation mechanisms. If necessary, also the non-destructive tests, laboratory tests (e.g. aggressivity of underground water) or material tests, such as the institute of witness samples or investigation of the boric acid influence on the concrete are used.

AMP Monitoring of Dukovany NPP Buildings [114] (the reactor building with the containment is a part of this Programme).

This Programme fulfils the role of the overall programme in the structural part. It includes the results of all Ageing Management Programmes and, what is more, the information from the in-service tests and inspections. Thanks to that, the Programme ensures that all available information is evaluated by a single expert at a single point.

The "Monitoring of Buildings" AMP serves as a tool for drawing-up a comprehensive overview of the physical condition of individual civil structures. It specifies the regular collection of all relevant information which was created for each civil structure in the given year and the output is the annual "Final Evaluation Report" for individual civil structures, which summarizes the results of individual tests, performed modifications and maintenance. Thanks to that, it enables monitoring of the changing condition of civil structures and their parts over time as well as the monitoring of compliance with the required functions.

AMP for Monitoring of Structures of the Dukovany NPP [115] (reactor building with the containment is a part of this Programme)

The subject of this Programme is the ageing management of selected civil structures. AMP also serves as a tool for defining the trends of the development of physical condition of the civil structure and its structures from the viewpoint of meeting their functions. The Programme is divided into two phases: The first phase is a visual inspection to identify the current state of civil structures. It makes it possible to identify the points with the degradation signs and possible sources of degradation. The output of the first phase is "Civil Structure Passport". Based on the evaluation of the current condition of civil structures, it is possible to suggest the corrective action or to continue with the second phase. The second phase is the so-called detailed survey and its output is the set of laboratory measurements, on-site measurements, static calculations, etc. with the comments.

Within this Programme, the following components are monitored:

- Foundations
- Underground structures
- Reinforced concrete structures – interior
- Reinforced concrete structures – exterior
- Finishing – coatings (for the purposes of surface decontaminability)
- Steel structures – interior
- Steel structures – exterior
- Hermetic steel liner
- Hermetic doors
- Steel liner (for the purposes of surface decontaminability)
- Stainless-steel liner of pools
- Constructional part of containment penetrations
- Fireproof seals
- Trapezoidal sheets
- Bolted joints
- Anchorage elements in the concrete
- Protective shielding caps (designed to be walked on or drive on)
- Foundation blocks and technology supports
- External cladding
- Roof cladding

AMP for Measuring of Civil Structure Settlement [116] (reactor building with the containment is a part of this Programme)

The Programme is used for measurement of settlement of civil structures and their parts due to the changes in the foundation soil under the structure or due to another building activity as a result of the static, dynamic or seismic load or other effects.

For the containment system, the ageing process is further elaborated in the Ageing Management Programmes focused on the condition of the hermetic zone structures and adjacent structures:

AMP for Containments in the Dukovany NPP [117]

The Programme applies to all containments in Dukovany NPP formed by the steam generator boxes, vacuum-bubbler condenser with air traps and the connecting corridor between the SG box and the vacuum-bubbler condenser.

The subject of the Ageing Management Programme for Dukovany NPP Containment is provision of the input data, see Chapter 7.1.3.1. The input data are processed and evaluated as per the requirements and criteria laid down in the specific Ageing Management Programme. The obtained results make it possible to monitor the development over time, i.e. the trend of changes in individual mechanical and physical characteristics, and to mitigate the effects of ageing by means of the introduction of early measures.

Within this Programme, the following components are monitored:

- Reinforced concrete structures of the hermetic zone
- Hermetic steel liners
- Hermetic hatches, closures and doors at the boundary and within the hermetic zone

AMP for Fuel Storage and Refuelling Pools in Dukovany NPP [118]

On each reactor unit, the subject of this Programme is the stainless steel liner which forms the inner surface of the Spent Fuel Storage Pool, Shaft No. 1, Decontamination Tank and the Storage Shaft for Active Equipment.

The subject of the Ageing Management Programme for Spent Fuel and Refuelling Pools is providing of input data as described in more detail in Chapter 7.1.3.1. The input data are processed and evaluated as per the requirements and criteria laid down in the specific Ageing Management Programme. The obtained results make it possible to monitor the development over time, i.e. the trend of changes in individual mechanical and physical characteristics, and to mitigate the effects of ageing by means of the introduction of early measures.

Within this Programme, the following components are monitored:

- Stainless steel liner forming the inner surface of the spent fuel pool
- Stainless steel liner forming the inner surface of the refuelling pool
- Stainless steel liner forming the inner surface of the Shaft No. 1
- Stainless steel liner forming the inner surface of the decontamination tank and the inner surface of the storage shaft for active equipment.

Scope of the AMP for the Temelín NPP

Generally, ageing of civil structures of the Temelín NPP is managed by means of the below Ageing Management Programmes (AMPs). These programmes help to monitor adverse effects of degradation mechanisms on the physical condition of civil structures and to predict the trend of future development. Thanks to that, the effective preventive or corrective actions can be taken to eliminate adverse effects of civil structures ageing and to ensure reliable and safe meeting of their design functions.

The Ageing Management Programmes and their development and general rules, principles and methods of the component selections are described in detail in Chapter 2 hereof.

<u>AMP for Civil Structures Parts of Containment in the Temelín NPP [119]</u>

The scope of the Programme applies to the containment of the Temelín NPP, parts of the penetrations built in containment structures, internals, transport corridor and hermetic closures.

The subject of the Ageing Management Programme for Component Parts of Containment in the Temelín NPP is providing of input data, see Chapter 7.1.3.1. The input data are processed and evaluated as per the requirements and criteria laid down in the specific Ageing Management Programme. The obtained results make it possible to monitor the development over time, i.e. the trend of changes in individual mechanical and physical characteristics, and to mitigate the effects of ageing by means of the introduction of early measures.

Within this Programme, the following components are monitored:

- Containment
    - Reinforced concrete structure of the hermetic boundary (cylindrical part, dome, bearing ring, foundation slab, instrumentation systems, roofing, painting of concrete structures from the outside)
    - Pre-stressing system (individual cables, anchors, cable ducts, protective grease for cables and anchors, protective covers and instrumentation system)
    - Steel lining of the hermetic boundary (hermetic steel lining of the containment and its secondary protection provided by painting from the inside)
    - Emergency boric acid storage pool (GA 201)

- Internals
    - Reinforced concrete structures (structures of the containment internals, horizontal and vertical structures of the transport corridor)
    - Steel structures (members supporting technological equipment inside the containment)
    - Steel lining (steel lining inside the internals of the containment and its secondary protection provided by painting from the inside)
    - Whip restraints

- Opening in the containment
    - Hermetic closures (hermetic hatch of the transport corridor, main and emergency hermetic airlock)

- penetrations

- Transport corridor
  - Reinforced concrete structures (horizontal and vertical structures of the transport corridor)
  - Steel lining (steel lining of the transport corridor and its secondary protection provided by painting from the inside)

- Openings in the transport corridor
  - Transport corridor door and two hermetic airlocks for personnel

AMP for Civil Structure Parts of Pools with Double Liner in the Temelín NPP [120]

The scope of the Programme applies to the pools with double liner. A system of pools with double liner is located inside the hermetic zone of the containment and the emergency boric acid storage pool is located at its boundary.

The subject of the Ageing Management Programme for Pools with a Double Liner is providing of input data as described in more detail in Chapter 7.1.3.1. The input data are processed and evaluated as per the requirements and criteria laid down in the specific Ageing Management Programme. The obtained results make it possible to monitor the development over time, i.e. the trend of changes in individual mechanical and physical characteristics, and to mitigate the effects of ageing by means of the introduction of early measures.

Within this Programme, the following components are monitored:

- Wet transport pool
- Inspection shaft for reactor internals and Inspection shaft for Control rod guide tube assemblies
- Reactor cavity
- Three sections of spent fuel storage pool
- Transport container shaft
- Boric acid solution pool
- Equipment decontamination shaft

AMP for Measuring of civil structure settlement [116] (reactor building with the containment is a part of this Programme)

The Programme is used for measurement of settlement of civil structures and their parts due to the changes in the foundation soil under the structure or due to another building activity as a result of the static, dynamic or seismic load or other effects.

Procedures for the identification of degradation mechanisms for the different materials and components of the Dukovany NPP

Degradation mechanisms affecting the individual components of civil structures have been identified according to the generalised worldwide experience and are based on the documents of US NRC GALL, IGALL Safety Report, EPRI, IAEA and ACI.

- US NRC NUREG – 1801, Rev. 2, Generic Ageing Lessons Learned (GALL) Report, 2010 [40]

- IAEA – Ageing Management for Nuclear Power Plants – International Generic Ageing Lessons Learned (IGALL), WIEN 2014 [22]
- EPRI Technical Report, Augmented Containment Inspection and Monitoring Report, 2013 [44]
- IAEA-TECDOC-1025, 1998 [41]
- IAEA NP-T-3.5, Ageing Management of Concrete Structures in Nuclear Power Plants, Vienna, 2016 [120]
- ACI 349.3R-02, Evaluation of Existing Nuclear Safety-Related Concrete Structures, Ronald J. Janowiak a spol, 2002 [42]

7.1.1.2 Scope of the Ageing Management Programme for reactor hall of the LVR-15 nuclear research reactor

The LVR-15 reactor is not equipped with protective pressure shell (containment) as required for nuclear power plants.

The overall design of the facility is based on the Soviet type project adapted to the national conditions. The composition of the reactor building as a whole is designed in the form of two connected parts - reactor hall and adjacent laboratory bay. The reactor hall comprises of a steel structure, inside which the reactor is located in a concrete shaft, which provides both protective and biological shielding function. The substructure is of monolithic design made from reinforced concrete. The reactor is based on concrete rafts; the foundations beneath the internal structure of the hall are made from reinforced concrete Be 170; the basement perimeter structures of the hall are made from reinforced concrete; the interior supporting structures are made from concrete (concrete Bd 135, Be 170 or heavy weight concrete with density of 3200 kg/m^3 - 4200 kg/m^3). The infilling structures of the hall between the steel beams above the level of $\pm$ 0.0 m consist of brick masonry made of P 100 bricks and cement-lime mortar type 25. The hall roofing is made of steel structure of the hall roof, with the TEBET prefabricated boards and the roof covering laid on it.

The steel structure of the hall is anchored in vertical perimeter structures. The backfill around the outside perimeter of the steel structure is pulled together by three horizontal reinforced concrete rings, hinge-connected to the steel structure, thus enabling the co-action of the backfill and the steel structure, in particular under the effects of horizontal pressure and wind suction. Structural system: welded steel poles hinged in bearings at elevation of $\pm$0.00 m, with crane runway girders at the level of rail of + 14.0 m. The hall roofing is made of truss girders with the span of 21.0 m. The solidity of the upper and lower rafts of the girder is ensured by longitudinal vertical ties. In the transverse direction, the acting forces are transferred by a longitudinal roof truss tie into the vertical truss ties located in gable walls. The steel supporting structure of the hall has its individual parts shop-welded; the assembly joints are partially riveted, partially screwed. The original design of steel structures and the static assessment are prepared according to the ČSN 05 0110 – 1949. – Design of steel structures for building construction.

The LVR-15 reactor object is shown in Figures A.16 and A.16 of Annnex A hereof.

7.1.2 Ageing assessment of reinforced concrete containments

7.1.2.1 Ageing assessment for reinforced concrete containments of the Dukovany and Temelín Nuclear Power Plants

As mentioned in Chapter 7.1.1.1, ageing management concerning the structures of both nuclear power plants is conducted according to the set of Specific Ageing Management Programmes. These programmes are periodically updated in the light of internal and external feedback and the current state-of-the-art in this field.

The following documents were used for the setting of the appropriate Ageing Management Programmes:

<u>Codes, standards, guides, production documentation, drawing documentation:</u>

- Operational documentation of ČEZ, a.s.
- As-built documentation
- ČEZ_ME_0870_Development of the Ageing Management Programme [28]
- ČSN ISO 13822 [123]
- IAEA IGALL [22]
- IAEA-TECDOC-1025 [41]
- IAEA NS-G-2.12 [6]
- INPO AP-913 [6]

<u>Use of operating experience</u>

- International guides of the EPRI, IGALL, ACCEPT
- Operating feedback taken into account – remedial actions

<u>Data from structural, production and assembly documentation</u>

- Operational documentation
- Structural drawings
- Chemical composition of materials
- Mechanical properties of materials
- Assumptions of project reviews

<u>Outputs of the research programmes used for the setting of the AMP for the Dukovany NPP:</u>

- Research programme - Boric acid influence on the concrete
- Long-term monitoring of load-bearing structural concretes around the Dukovany NPP reactor cavity, Technical University Brno
- Evaluation of the measurements carried out during integrity verification test of Dukovany NPP Unit 1, Technical University Brno

<u>Outputs of the research programmes used for the setting of the AMP for the Temelín NPP:</u>

- Moisture monitoring of reinforced concrete of internal structures of containment
- Laboratory corrosion resistance testing for concrete and reinforcement in water solutions of boric acid, implemented by ÚJV Řež, a.s.

<u>The other inputs for setting the AMP were as follows:</u>

- Data and results of the history of containment operation load (data concerning the transient operating modes)
  - Parameter (temperature, pressure) patterns pertaining to the individual transient operating modes

- Data and results of the factory, erection and in-service inspections
    - Results of the factory, erection and pre-service non-destructive testing, the so-called "map of the indications found on the containment", which provides an overview of all the indications found, stating the type, size and place of the indication
    - Information concerning the scope of containment inspection and the inspection procedure
- Data and results of the inspections currently being carried out

The information set out was used for:

- The identification of individual degradation mechanisms, the determination of their effects and their assignment to structural components (e.g. chemical corrosion of concrete is indicated by the development of extracts, blooms or cracks on the surface of the structure)
- The assessment of significance of and need for the management of degradation mechanisms
  (e.g. the mechanisms related to the effects of seawater were completely excluded; less attention was paid to unlikely mechanisms such as alkali-silica reaction (ASR) of aggregate and attention was focused on usual and the most acting mechanisms such as concrete degradation due to the aggressivity of surrounding medium or cracks developed due to creeping, settlement, etc.)
- The setting of acceptance criteria, pertaining to the effects of individual degradation mechanisms (e.g. criteria for allowable crack thickness according to the applicable Eurocodes or criteria for allowable extent of extracts or blooms on the surface of the structure were established, criteria for the minimum level of pre-stress were established, etc.)

The output of the ageing assessment of structures is the annual final assessment report for individual civil structures including proposal for the measures to be taken to ensure functionality and measures supporting long-term operation (the conclusions are then part of the Health Reports, thus serving as feedback and information concerning the effectiveness and accuracy of maintenance).

In addition, the annual final report is drawn up in the Temelín NPP for the containment, with the graphs with extrapolation of the expected decrease in pre-stressing force and proposal for the measures to be taken to ensure functionality and measures supporting long-term operation.

Identified degradation mechanisms and ageing effects including identification of their significance

Structures are typically made from several building materials. Degradation mechanisms are divided into groups for individual materials.

Identical potential and real degradation mechanisms have been identified for both types of plants. These can be divided by their effects on individual materials as follows:

<u>Concrete:</u>

- Calcium hydroxide leaching and efflorescence
- Sulphate corrosion

- Acid or alkaline corrosion
- Alkali-silica reaction (ASR)
- Carbonatation
- Crystallization of chlorides and other salts
- Freeze-thaw
- Abrasion, erosion, cavitation
- Fatigue, vibrations
- Overload
- Microbiologically induced corrosion
- Settlement

<u>Steel:</u>

- Corrosion in general
- Stress corrosion cracking
- Fatigue/fatigue fracture
- Overload
- Settlement
- Microbiologically induced corrosion
- Abrasion
- Chemical degradation

<u>Reinforced-concrete structures:</u>

- Cracks
- Leaching, efflorescence, seepage
- Flaking concrete
- Erosion, abrasion
- Spalling concrete
- Concrete disintegration

<u>Pre-stressing system (only relevant to the pre-stressed containment of the Temelín NPP)</u>

- Loss of pre-stressing due to relaxation of steel, concrete creep and increased temperature
- Material losses on pre-stressing cables – general, pitting, slot corrosion
- Material fatigue

Degradation mechanisms specifically addressed under the AMPs for containments:

<u>Concrete:</u>

- Boric acid corrosion
- Increased temperature, temperature cycles, concrete water loss
- Radiation damage

<u>Hermetic liner steel – hermetic liner thickness:</u>

- Corrosion in general
- Microbiologically induced corrosion
- Chemical degradation

<u>Steel for hermetic closures and doors:</u>

- Corrosion in general
- Wear

Pre-stressing system (only relevant to the pre-stressed containment of the Temelín NPP)

- Loss of pre-stressing due to relaxation of steel, concrete creeping and increased temperature
- Material losses on pre-stressing cables – general, pitting, crevice corrosion
- Material fatigue

Degradation mechanisms specifically addressed under the AMPs for Fuel Storage and Refuelling Pools:

Stainless steel:

- Corrosion in general
- Intergranular stress corrosion cracking (IGSCC)
- Transgranular stress corrosion cracking (TGSCC)
- Fatigue / fatigue fracture
- Microbiologically induced corrosion
- Abrasion
- Chemical degradation

7.1.2.2 Ageing assessment for reactor hall of the LVR-15 nuclear research reactor

In the original design of structures the emphasis is placed on robustness with regard to erection and operation load while, when viewed from today's perspective, the issue of the robustness in external extraordinary events wasn't fully appreciated.

Therefore, a number of analyses were prepared in 1996 to verify the full compliance of the existing buildings with the existing legislation of the Czech Republic, technical standards, IAEA regulations and the international accepted practices for the assessment of research reactors.

The methodology for assessing the robustness of existing structures, which contains the calculation principles, loading rules and an overview of technical standards and regulations acceptable to the assessment of safety relevant structures associated with the operation of the LVR-15 reactor, was developed.

The analysis itself and the classification of extraordinary external events to be taken into account in the given site were undertaken as part of the "Assessment of the robustness of the LVR-15 research reactor and the high-level waste storage in extraordinary external effects (stage 1 - analysis of the individual effects and the need for taking them into account in robustness assessment of the needs for Final Safety Analysis Report) Stevenson and Associates, July 1996". Following the analyses carried out, recalculations were performed for all structures.

The assessment of concrete and steel structures is currently based on the so-called "limit state design method", which permits, compared to older approaches, a more concise expression of the level of uncertainty of input parameters, thus ensuring a higher level of reliability.

Since the 1950s, when the load bearing structure was implemented, a substantial tightening of the requirements for nuclear safety and significant changes in the requirements for the level of reliability and safety of nuclear installations were made. At the time of design origination, the issue of external extreme load of both natural (seismicity, floods, climate extremes) and human-induced (industry, external explosions, aviation accidents) loads has been particularly underrated.

Therefore, new analyses "Assessment of the robustness of civil structures of the LVR-15 reactor building (building no. 211/1) and its annex buildings and the high-level waste storage (building no. 211/08) in ÚJV Řež, a. s. in extraordinary external effects" were prepared in 1996 to verify the real robustness of buildings in the light of new standards, regulations and according to the internationaly accepted practices.

Furthermore, analyses were carried out in 1996 to verify compliance with the current legislative requirements with respect to the robustness of the structures associated with the operation of the LVR-15 reactor. The original calculations from the 1950s were carried out as simplified because the methods and equipment known at the time did not allow detailed analyses of the mechanical behaviour of more complex spatial systems. Spatial computational models were developed for the analysis of reactor building and associated structures.

The analysis of civil structures was carried out for all load combinations.

The methodology for carrying out analyses, developing computational models and principles for assessment is stated in the report "Methodology for assessing the robustness of the LVR-15 research reactor (building no. 211/1 and associated buildings) and the high-level waste storage (building no. 211/8) in ÚJV Řež, a. s. in an earthquake and other extraordinary external events, Stevenson and Associates, September 1996".

The methodology for assessing seismic effects is particularly detailed.

The important factor for the assessment of civil structures for the effects of external extremes with a low probability is that the partial damage to buildings and permanent deformations may be accepted, and only the collapse of global load bearing structure should be prevented while maintaining the safety function of reactor containment.

Robustness assessment of this structure was carried out in accordance with the methodology by calculation, i.e. the CDFM method. The results are available in the report "Assessment of the robustness of civil structures of the LVR-15 reactor building (building no. 211/1) and its annex buildings and the high-level waste storage (building no. 211/08) in ÚJV Řež, a.s. in extraordinary external effects, Stevenson and Associates, September 1996".

The post-Fukushima stress tests on the LVR-15 reactor resulted in the requirement for the assessment and evaluation of robustness against extreme and very unlikely phenomena. In this assessment of the LVR-15 reactor it was stated that the design basis earthquake (according to the Safety Guides 50-SG-S1 and 50-SG2-S2) with the peak ground acceleration PGA = 0.05 cannot lead to the overall collapse of the structure or such damage that would compromise the performance of safety functions of research reactor equipment. Such an earthquake may cause only local damage to structures, rupture in masonry, falling plaster, broken glass lining in the reactor hall.

7.1.3 Monitoring, testing, sampling and inspection activities for containments

7.1.3.1 Monitoring, testing, sampling and inspection activities for containments of the Dukovany and Temelín Nuclear Power Plants

Activities under the Ageing Management Programmes in the Dukovany NPP concerning the individual parts of civil structures:

<u>Hermetic steel liners</u>

- Hermetic liner thickness
  - The phased array ultrasonic method is used - diagnostics of the areas of carbon liners, focusing particularly on areas potentially significantly reduced due to corrosion.

    - Pilot measurement takes place gradually on individual units; suitable testing period will be determined at the end of measurement of all units. The minimum allowable thickness of liner is determined.
- Overall leakage from the hermetic zone during periodic integral tightness testing (PERIZ)
    - Measurement of the leakage from the hermetic zone during tightness test under reduced overpressure of 50 kPa with conversion to the leakage under design basis overpressure of 150 kPa by means of extrapolation coefficients (updated for each unit).
    - By default, the measurement is carried out every two years (provided that the result of the previous test is not worse than 30% or provided that there is no significant intervention in the containment system). The maximum allowable leakage shall not exceed 13%wt/24 hours.
- Leakage from the hermetic zone during tightness test through the drainage of controlled leakage system from the spent fuel pool and refuelling pool
    - Measurement of the leakage from the hermetic zone through the hermetic liner into the area of the system for collecting controlled leakage from the spent fuel pool and refuelling pool. The proportion of leakage through this part of inaccessible hermetic liner in the overall leakage of hermetic zone is determined
    - The measurement is carried out every two years. The proportion in overall maximum leakage is determined and the leakage through the controlled leakage system is part of "The maximum allowable leakage", which shall not exceed 13%wt/24 hours.

<u>Hermetic covers, closures and doors at the boundary and within the hermetic area</u>

- Hermetic hatches and doors at the boundary and within the hermetic area are subject to local tightness tests. The closing of the door should be satisfactory and should be confirmed in the satisfactory report of local tightness test. Otherwise, measures are implemented to ensure the satisfactory closing of the door (for example, replacement of seal).
- They are carried out annually at the end of outage.

Description of the acquisition of data to meet the AMP for Fuel Storage and Refuelling Pools in the Dukovany NPP

<u>Spent fuel storage pool</u>

The following parameters are monitored

- Monitoring of the amount of the leak detected to the controlled leakage system from the spent fuel storage pools (once a month - if necessary, more often)
    - In case of leakage detection in stainless steel liner into the intermediate space, the leak detected is removed from the section by the drainage system into the collecting containers, where it is on-line monitored by the automatic local water level measurement system.

    - Data containing information about:
        - Localisation of the sections with leakage
        - Amount of leaks
        - Continuity of leaks
- Non-integrity, damage to stainless steel liner surface above the water level through cracks
    - Monitoring of the condition of the accessible surfaces of inner stainless steel pool liner causing integrity failure by cracks
    - It is carried out once a year
- Non-integrity, mechanical damage to stainless steel liner surface above the water level
    - Monitoring of the condition of the accessible surfaces of inner stainless steel pool liner causing integrity failure by mechanical damage (with emphasis on welded joints) caused, for example, by penetration during handling of devices to be inserted
    - It is carried out once a year
- Surface condition of stainless steel pool liner for corrosion
    - Occurrence of stainless steel liner corrosion on visible surfaces is monitored
    - It is carried out once a year
- Leaching, salt deposits, moisture
    - Monitoring of the occurrence of leaching, salt deposits and moisture
    - It is carried out once a year
- Number of filling and emptying cycles
    - Absolute number of cycles over the life of NPP,
    - Number of cycles in a given calendar year
    - Flooding period of the areas to the maximum water level during exchange of fuel rods
    - It is evaluated once a year
- Solution temperature pattern
    - Maximum and minimum absolute temperature of solution
    - Rate of change in solution temperature
    - It is evaluated once a year
- Mechanical deposits
    - Monitoring for the occurrence of mechanical deposits
    - It is carried out once a year

The acceptance criteria for spent fuel storage pool are based on the following:

- amount of leaks in litres per day, determined for individual water levels
    - 160 litres/day at the water level of 14.45 m
    - 270 litres/day at the water level of 18.5 m (upper grate filling)
    - 335 litres/day at the water level of 21 m (refuelling)
- increase in leakage compared to the previous monitored period – in percentage

Shaft No. 1

The following parameters are monitored

- Amount of the leak detected into the controlled leakage system - on-line
- Non-integrity, damage to stainless steel liner surface above the level through cracks or mechanically - once a year
- Occurrence of stainless steel liner corrosion on visible surfaces - once a year
- Leaching, salt deposits, moisture – once a year
- Number of filling/emptying cycles - once a year
- Mechanical deposits - once a year

The acceptance criteria for spent fuel storage pool are based on the following:

- amount of leaks in litres per day, determined for individual water levels
    - 60 litres/day at the water level of 14.6 m
    - 125 litres/day with full filling
- increase in leakage compared to the previous monitored period – in percentage

Decontamination tanks and storage shafts for active equipment

The following parameters are monitored

- Non-integrity, damage to stainless steel liner surface - once a year
- Occurrence of stainless steel liner corrosion on visible surfaces - once a year
- Leaching, salt deposits, moisture – once a year

Beyond the above activities included into the scope of different programmes of the plants (AMP, In-service Inspection Programme, LaC, etc.), the following inspection, scientific and research activities are carried out:

- Visual inspection and subsequent passportization of civil structures
    - A visual inspection shall be carried out by an assessor aimed at identifying symptoms of degradation mechanisms on individual structures/components for the selected civil structures
    - During inspection, the individual structures/components and their constituent materials are assessed in terms of effects of the possible degradation mechanisms referred to in the AMP for Monitoring of Structures of the Dukovany NPP
    - The passportization of the Reactor Building 800/1 is carried out on a yearly basis; the passportization of other structures is carried out at an interval of 1 to 3 years depending on their condition and the trend of the development of their condition

- Temperature patterns - Temperature measurement sensors are located in reactor cavities. They are in the areas, where the temperature reaches its highest values. Temperature measurement in other parts of the reinforced concrete structures of hermetic zone does not need to be performed. The evaluation is carried out once a year.

The following quantities are measured:

- Nitrogen temperature in the channels of ionisation chambers
- Dry protection concrete temperature
- Concrete bracket temperature
- Support frame temperature
- Concrete temperature in reactor cavity

- Deformation of the roof and wall of bubblercondenser tower
  Deformation measurement of the roof and wall of bubblercondenser tower by means of the HPL method - High Precision Levelling, and internal continuous measurements during tightness test under reduced overpressure of 50 kPa with the use of the existing extrapolation for deformation under design basis overpressure of 150 kPa by means of the numerical model. Measurement and evaluation are carried out in a period of two years.

- Institute of Concrete Surveillance Specimens (implemented according to the time schedule for the assessment of surveillance specimens).
  In the framework of safety and reliability review of the Dukovany NPP, long-term quality monitoring of shielding and load bearing concretes in the surroundings of the reactor was initiated in order to acquire knowledge about the impact of nuclear radiation, increased temperature and moisture on their mechanical-physical parameters. The quality of shielding concretes and concrete structures in the surroundings of the reactor was not controlled in this way. And that is why the system has been designed in the Dukovany NPP for quality control of concretes exposed to the effects of radiation, temperature and moisture load with the use of the method of surveillance specimens. The objective of the long-term monitoring of surveillance specimens including the existing concretes in the area around the Dukovany NPP Unit 1 and 3 reactor cavity is to verify their behaviour and to identify any changes in physical-mechanical characteristics and chemical composition induced by temperature, moisture, mechanical and radiation load. The obtained data can be used for assessing the structures for all the specified types of load and their combinations that the structures of nuclear power plant should withstand.
  The following parameters are assessed:
  - Neutron fluence
  - Gamma radiation
  - Compressive strength of concrete
  - Content of artificial radionuclides
  - Density
  - Mineral composition
  - Borate content

- Research programme - Boric acid influence on the concrete

    - In cooperation with ÚJV Řež, a.s., the specimens of structural concrete and hermetic liner including its surface finish are being exposed to the effects of boric acid solution takes place
    - It was carried out according to the Draft Test Programme, the Assessment Report after 1, 2, 3, 5, 7, 10, 15, 20, 25 years of exposure

Activities under the Ageing Management Programmes in the Temelín NPP:

Description of the acquisition of data to meet the AMP for Component Parts of Containment in the Temelín NPP

- Pre-stressing force test
    - Hydraulic presses (Lift-off test)
    - Pre-stressing force in a cable by the tensometric measuring system (Hottinger foil-based resistance strain gauges affixed to the anchor bolts in tension on the ring at the point of contact with the orbit)
    - Pre-stressing force at specified points along the length of a cable by the MEM measuring system (magneto-elastic method, dependence of the magnetization of ferromagnetic material on the mechanical stress)
- Structural response to load (component of the reinforced concrete structure of hermetic boundary)
    - Deformation of concrete structures (strain) by PLDS and SDM-B sensors (stringed strain gauges)
    - Force in concrete reinforcement by PSAS and SDM-V sensors (stringed strain gauges)
    - Displacements of structures by PLPS sensors (stringed strain gauges) and geodetic surveying
    - Structural temperature by PTS and SDM-T sensors
- Condition of pre-stressing system
    - Inspection of anchor covers
        - Visual inspection of anchor covers
        - Visual inspection of gasket/seal (only for dismantled covers)
        - Condition of rubber gaskets/seals
- Moisture test
    - Test at the point of anchors (moisture at the point of anchors and cable ducts - long-term moisture monitoring)
    - Moisture in cable ducts (range of present moisture)
- Control of protective grease
    - Condition and integrity of protective grease on anchors and pre-stressing cables
- Inspection of cables for mechanical damage and corrosion
    - Mechanical damage to the wire and anchoring device (cracks, abrasion, scratching, wire breaking, wire crossing, etc.)
- Removal and inspection of selected cables
    - This includes a set of inspections and laboratory tests on cable samples

- Condition of reinforced concrete (reinforced concrete structure of hermetic boundary)
  - Visual inspection of concrete surface + compensator area
    - Condition of concrete in accessible areas, focusing on inspection windows
    - Condition within the anchoring blocks of pre-stressing cables (possible clogging of drains of atmospheric water in the area of cylindrical anchor cables)
    - Condition of coatings on the outer surface of concrete structures by visual inspection
- Condition of the cover of dome, cornice and bearing ring
  - Condition of the Vulkem insulation – dome + cornice (for Unit 1)
  - Condition of PVC insulation (for Unit 2)
  - Condition of the insulation coating of the cornice (inspection carried out for Unit 2)
  - Condition of sealing compound for joints between the edge panels of cornice
  - Condition of atmospheric water drains in the area of anchoring cylindrical cables
- Inspection of the development of cracks in concrete
  - Origination and development of cracks in concrete, drawing up and updating of a map of cracks
- Reinforced concrete testing (carried out on the structure without sampling)
  - Concrete strength detected by NDT method
  - Concrete carbonation
  - Corrosion inspection of concrete reinforcement
- Inspection of the foundation part of containment (reinforced concrete structures of the hermetic boundary)
  - Condition of the coating system by visual inspection
  - Visual condition of the surface of steel lining (mechanical damage, rust stains, etc.)
- Condition of the hermetic steel lining of containment
  - Visual inspection of the coating system of steel lining
    - Condition of the coating system of steel lining of the containment by visual inspection
- Visual inspection of the condition of steel lining and cavities
  - Visual condition of the surface of steel lining (mechanical damage, corrosion, leaching and salt deposits, cracks, etc.)
  - Deflection of steel lining from concrete structure by acoustic routing and drawing up a map of cavities
  - Non-destructive thickness measurement of steel lining
    - Steel lining thickness detected by NDT methods (with ultrasound in the designated inspection points)
- Condition of the steel lining of internals
  - Condition of the coating system of steel lining of containment internals by visual inspection and thickness of coating
  - Visual inspection of the condition of steel lining
    - Visual condition of steel lining - mechanical damage, rust stains, etc.

        - Non-destructive thickness measurement of steel lining
            - Steel lining thickness detected by NDT methods
- Condition of the hermetic steel lining of the transport corridor
    - Condition of the coating system of steel lining of the transport corridor by visual inspection and thickness of coating
    - Visual inspection of the condition of steel lining of transport corridor
    - Non-destructive thickness measurement of steel lining of the transport corridor
        - Use of NDT method – ultrasonic thickness testing of steel lining at inspection points (always at the same points)
- Condition of the hermetic closures of containment
    - Condition of the rubber seal/gasket of hermetic closures
        - Visual inspection
        - Non-destructive hardness testing of sealing rubber
    - Visual condition of the surface of hermetic closure
        - Surface inspection
        - Mechanical damage
    - Functionality
        - Functionality of mechanisms
        - Functionality of electrical connections
        - Number of openings of the closure
- Condition of the hermetic closures of transport corridor
    - Condition of the rubber seal/gasket of hermetic closures
        - Visual inspection of seals/gaskets
        - Non-destructive hardness testing of sealing rubber
    - Visual condition of the surface of hermetic closure
        - Surface inspection
        - Mechanical damage
    - Functionality
        - Functionality of mechanisms
        - Functionality of electrical connections
        - Number of openings of the closure
- Condition of hermetic penetrations from the inside of the containment
    - Visual inspection of the coating system
        - Condition of the coating system of hermetic penetrations of the containment by visual inspection (cracks, peeling coat, mechanical damage, coat discolouration)
    - Visual inspection of the condition of steel surface
        - Visual condition of steel surface of hermetic penetrations (mechanical damage, corrosion, leaching and salt deposits, cracks)
- Condition of building hermetic penetrations from the outside of the containment
    - Visual inspection of the coating system
        - Condition of the coating system of hermetic penetrations of the containment by visual inspection (cracks, peeling coat, mechanical damage to the coat, coat discolouration)

    - Visual inspection of the condition of steel surface
        - Visual condition of steel surface of hermetic penetrations (mechanical damage, corrosion, leaching and salt deposits, cracks)
- Overall tightness of the containment
    - Leak-tightness of the whole area of containment
        - PERZIK – Periodic integrity test of the containment (tightness test, leakage from the area of containment)
- Local tightness
    - Tightness of selected nozzles and cells in steel lining of the containment
    - Tightness of selected nozzles and cells in steel lining of the transport corridor
    - Tightness of the penetrations in the structural part
    - Tightness of the main hermetic airlock
    - Tightness of the emergency hermetic airlock
    - Tightness of the hermetic hatch
    - Tightness of the doors of transport corridor
    - Tightness of the hermetic hatch of the transport corridor
- Hatch Analysis of groundwater test bore holes
    - Laboratory analysis of groundwater
        - The amount of undesirable elements of degrading reinforce concrete structure
        - Determination of groundwater pH
- Condition of internals concretes
    - Moisture monitoring of internals concretes
        - Concrete moisture at each individual measurement point
    - Chemical composition analysis of internals concrete
        - Chemical composition of concrete
        - Sulphate corrosion in concrete
        - Ongoing alkali-silica reaction
    - Tests of the impact of boric acid solution on internals concrete
        - Impact on structural concrete
        - Impact on heavy weight concrete

<u>Description of the acquisition of data to meet the civil structures parts AMP for the Temelín NPP pools with double liner</u>

- Visual inspection of austenitic pool liners
    - Condition of the accessible surfaces of inner austenitic pool liner causing integrity failure by cracks (monitoring for potential mechanical damage)
    - Condition of the accessible surfaces of inner austenitic pool liner causing integrity failure by mechanical damage (with emphasis on welded joints) caused, for example, by penetration during handling of devices to be inserted (monitoring for potential mechanical damage)
    - Condition of the surfaces of inner austenitic pool liner for corrosion, in particular for hollow welds (monitoring for potential corrosion to austenitic liner)

- Condition of the surfaces of inner austenitic pool liner for leaching and deposits of boric acid salts due to solution discharges from the intermediate space into the pool (monitoring of blooms and discharges)
- State of the presence of mechanical deposits

- Monitoring of pool filling cycles or water level changes
  - Absolute number of cycles over the life of NPP
  - Frequency of the cycles in each year
  - Flooding period of the areas to the level +36.20 during exchange of fuel rods
- Solution temperature monitoring of permanently flooded areas over time (data from periodic measurements in the periodicity according to the In-service Inspection Programme)
- Detection and monitoring of seepage through the liner of all pools outside the boron pool into the intermediate space of liners
  - Whether or not there is any seepage
  - Recording of the amount of discharges into the intermediate space between the two shells into the reservoirs of the TZ50 system
  - Localisation of leakage of liquid – particular sections
  - Whether the leak is continuous or random
  - Chemical analysis of any liquid leaked following each outage
  - Amount of any liquid leaked from the section in litres/day (limits are determined)
- Detection and monitoring of seepage through the liner of boron pool into the intermediate space of liners
  - Whether or not there is any seepage
  - Recording of the amount of leakage into the intermediate space between inner and outer liner
  - Localisation of leakage of liquid – particular sections
  - Absolute amount of solution leaked – in litres
  - Whether the leak is continuous or random
- Monitoring of changes in the amount of the solution refilled for each refilling
- Data concerning the location of spent fuel rods over time and spent fuel storage pool in relation to thermal stress of liners
  - Monitoring of the spatial location of spent fuel rods over time and spent fuel storage pool in relation to thermal stress of liners

7.1.3.2 Monitoring, testing, sampling and inspection activities for structures of the LVR-15 nuclear research reactor

The structures are not included within the scope of the Ageing Management Programme of the LVR-15 nuclear research reactor; however, they are monitored under the In-service Inspection Programme.

Building construction is visually inspected in order to detect any potential damage to wall surface – cracks, in a period of once a year. Furthermore, settlement of the building is checked at an interval of every five years.

7.1.4 Preventive and remedial actions for reinforced concrete containments

7.1.4.1 Preventive and remedial actions for reinforced concrete containments of the Dukovany and Temelín Nuclear Power Plants

Timely indication of undesirable trend of containment material or component degradation is the prevention of ageing effects or serves as a basis for developing the activities to mitigate ageing. In the case of any indication, the proposal for mitigating measures follows, which can differ for each case.

Remedial actions are described in more detail in individual AMPs. The procedure for remedial action is provided in the AMP for each monitored parameter.

For example, in the case of detecting corrosion in hermetic liner, the in-situ measurement is carried out to verify the actual condition and extent of damage, the cause of corrosion is analysed (e.g. foreign material / impurity embedded in concrete or absence of the contact with passivating concrete in the presence of moisture, etc.). Maintenance of the function of weakened liner is assessed by calculation and remedial action is taken (keeping or replacing the damaged area depending on the result of calculation or additional real measures).

In the context of the preparation of the Dukovany NPP for LTO, the setting of existing processes and the individual documents of the Ageing Management Programmes were thoroughly reviewed.

The level of ageing management was compared to the worldwide practices by both operator's (ČEZ) personnel and SALTO mission, where staff members of the national regulators of the international operators of nuclear power plants were represented.

The two procedures independent of each other generated identical conclusions:

- Need for the review of the current Ageing Management Programmes to meet the qualitative requirements imposed thereon by international recommendations (e.g. IAEA NS-G-2.12 Ageing Management for Nuclear Power Plants)
- Need for extending the Ageing Management Programme to other important civil structures outside the containment itself and the spent fuel storage pools.

The recommendations were accepted and fulfilled in the course of 2016, which led to the extension of equipment condition inspections. Preventive actions are taken with regard to operating conditions, which have the effect on change in material properties of civil structures during their operation, specifically:

- Operation culture, i.e. compliance with operating procedures, Limits and Conditions

For spent fuel storage pools, the measures to mitigate ageing effects are particularly implemented with regard to the following:

- Operating conditions that affect the change in material properties of the pools with double liner during their operation, e.g.:
  - Layout of the spent fuel assemblies in the spent fuel storage pool so as to ensure continuous temperature distribution of solution and in pool walls
  - Compliance with the allowable solution temperature gradient when filling all pools in the exchange of fuel elements

- Not exceeding the maximum prescribed solution temperature during unit operation and during refuelling

7.1.4.1 Preventive and remedial actions for structures of the LVR-15 nuclear research reactor hall

Based on of the conclusions of the Summary Report “Assessment of the robustness of the LVR-15 research reactor and the high-level waste storage in ÚJV Řež, a.s. in an earthquake and other external extraordinary initiating events, Stevenson and Associates, September 1996”, which involves the potential for external flooding of the reactor building, it was recommended to implement site measures to prevent the ingress of water into the building in extreme floods. After floods in 2002, the ÚJVTechnical Section implemented a number of measures to prevent/mitigate the consequences of any potential flood.

In addition, the In-service Inspection Programme for the LVR-15 reactor was completed with the visual testing of changes of bedrock volume and any origination and development of the cracks occurred due to uneven settlement, caused by changes in groundwater level as a result of floods. Inspections under the In-service Inspection Programme with a yearly interval are periodically carried out by a specialist for structural statics and dynamics (Energoprojekt Praha).

7.2 Operator’s experience of the implementation of the AMPs for reinforced concrete containments

7.2.1 Dukovany and Temelín Nuclear Power Plant operator’s experience of the implementation of the AMPs for reinforced concrete containments:

Dukovany NPP

- *AMP for Monitoring of Buildings* - The annual “Final Assessment Reports” for each civil structure will be drawn up since 2017
- AMP for Monitoring of Structures of the Dukovany NPP - after the introduction of the AMP for Monitoring of Buildings (passportization), the development of the physical condition of civil structures and individual structures in the Dukovany NPP will be systematically monitored. This process started in early 2017 for reactor units 3 and 4, and will be put into practice for reactor units 1 and 2 in 2018
- *AMP Measuring of Civil Structure Settlement* (reactor building with the containment is a part of this Programme). The course of settlement of Unit 1 corresponds to the theoretical assumptions where the subsurface consolidated at the first stage following construction due to additional load by construction and the trend of settlement subsequently slowed down or there is no longer any further settlement. The values of settlement are far below the maximum settlement limit, as determined according to ČSN 73 1001. For settlement measurement of Unit 2, the individual measuring points show very small movements and therefore the maximum settlement limit is not exceeded. It can be concluded that the results of long-term settlement monitoring of Unit 1 and Unit 2 confirm the smooth functioning of the foundation of these structures.
- *AMP for Containments in the Dukovany NPP*

- Hermetic steel liners
  - *Hermetic liner thickness* - from 2017, steel hermetic liner thickness will be measured with the use of the phased array ultrasonic method. The operating procedure and the measuring points are defined in Annex to the AMP for Containments in the Dukovany NPP
  - *Overall leakage from the hermetic zone during periodic integral tightness testing (PERIZ)*

    The results of the pressure tests carried out so far show compliance with the allowable amount of containment leakage with a sufficient reserve, thus not exceeding the maximum allowable leakage of 13%wt/24 hours.
  - *Leakage from the hermetic zone during tightness test through the TZ system – drainage of controlled leakage from the spent fuel pool and refuelling pool.*

    The trend of overall leakage from pool and shafts no. 1 on Units 1 to 4 is "improving" or "sustained"
- Hermetic hatches, closures and doors at the boundary and within the hermetic zone area

  Tightness of these hermetic parts is verified during local tightness tests. Possible defects are immediately removed and the result of the local tightness test must be test protocol of satisfactory status.
- Temperature patterns

  The measured temperature values are below the maximum limits in individual parts of the reactor shaft
- Deformation of the roof and wall of the bubblercondenser tower

  The maximum deformations measured during containment tightness and integrity testing are below the value determined for maximum structural deformation
- Institute of Concrete Surveillance Specimens (implemented according to the time schedule for the assessment of surveillance specimens)

  The results of long-term monitoring of the Institute of Surveillance Specimens confirm that radiation has not yet any impact on dynamic characteristics of concrete. The effect of different moisture in concrete sample was observed
- Research programme - Boric acid influence on the concrete

  In 2016, the programme was launched for testing boric acid influence on the concrete. In order to ensure the best possible values, the samples of concrete were manufactured in conformity with the formulation used in the design of the Dukovany NPP.

The results of the assessment show that no limit value for the monitored parameters is exceeded.

The existing condition of the civil structure with regard to the level of the sustainable functional properties required for long-term operation of the Dukovany NPP for a minimum period of next ten years can be assessed as "Acceptable".

- *AMPs for Fuel Storage and Refuelling Pools in the Dukovany NPP*
  - Spent fuel storage pool
    - Monitoring of the amount of the leak detected to the TZ controlled leakage system from the spent fuel storage pools
      The long-term analysis of automatic records shows that the amount of the leak detected into the TZ controlled leakage system does not exceed the limit value on a long-term basis.
    - Number of filling and emptying cycles
      The monitoring of pool filling/emptying cycles shows that the measured values are below the specified limit values
    - Solution temperature pattern
      Temperature in storage pools for prior periods as well as rate of change in solution temperature did not exceed the limit values.
    - Non-integrity, damage to stainless steel liner surface above the level through cracks or mechanically
      This parameter is monitored by means of visual inspections that will begin in the course of 2017 – 2018
    - Surface condition of stainless steel pool liner for corrosion
      This parameter is monitored by means of visual inspections that will begin in the course of 2017 – 2018
    - Mechanical deposits
      This parameter is monitored by means of visual inspections that will begin in the course of 2017 – 2018.

  - Shaft No. 1
    - Amount of the leak detected into the controlled leakage system
      The long-term analysis shows that the amount of the leak detected into the TZ controlled leakage system does not exceed the limit value on a long-term basis.
      The parameters that are monitored by means of visual inspections will be assessed in the course of 2017 – 2018.

  - Decontamination tanks and storage shafts for active equipment
    From 2017 – 2018, the monitoring of the above mentioned parameters (Chapter 7.1.3) will be carried out by means of visual inspections.

The results of the assessment show that no limit value for the monitored parameters is exceeded. There are therefore no restrictions known for plant operation.

Temelín NPP

- *AMP for Component Parts of Containment in the Temelín NPP*
  - Pre-stressing force test
    The evaluation of individual measuring systems showed compliance with the requirement for the minimum pre-stressing level with a sufficient reserve.
  - Structural response to load (component of the reinforced concrete structure of hermetic boundary)
    The measurement of structural response to the loads applied shows stabilised condition of the structure and the development of stressed-deformation condition of the structure corresponds to the expected development.
  - Condition of pre-stressing system
    The inspections of pre-stressing system carried out so far showed no defects or damage reducing the functionality of this system. The presence of water under anchor covers is solved by draining the covers and preparing the adaptations of covers (ventilation); the cause is solved by preparing the repairs of roof covering.
  - Condition of reinforced concrete (reinforced concrete structure of hermetic boundary)
    The visual inspections of the concrete surface of containment showed no defects that would reduce its functionality.
  - Condition of the cover of dome, cornice and bearing ring
    The visual inspections showed the need for the overall repair of roof covering of the dome of the containment and for the completion of repairs of water-proof insulation on the roof of enclosure due to the material degradation.
  - Inspection of the development of cracks in concrete
    The defects (cracks) detected in concrete surface have no effect on the current function of the structure, but each individual defect will be repaired.
  - Reinforced concrete testing (carried out on the structure without sampling)
    The results of non-destructive testing showed that compressive strength of concrete meets the requirement for the minimum guaranteed compressive strength of concrete of the concrete class used.
    Furthermore, the visual surface inspections demonstrate that there are no defects on the surface of the containment that would reduce the functionality of the structure.
  - Inspection of the foundation part of containment (reinforced concrete structures of the hermetic boundary)
    The visual inspections of the foundation part of containment showed no defects that would reduce its functionality.
  - Condition of the hermetic steel lining of containment
    The inspection of steel lining of the containment demonstrated that the steel lining is intact, without mechanical damage, without cracks in welds and base material including coating on it and according to the reports of visual inspection, its condition is evaluated as satisfactory.

The thickness of steel lining of the containment is within the tolerance required by the production standard for metallurgical products and is, as a whole, satisfactory.

- Condition of the steel lining of internals

  The inspection of steel lining of the internals (rooms inside the containment) demonstrated that the steel carbon lining is intact, without mechanical damage, without cracks in welds and base material including coating on it, and stainless steel lining is also without surface corrosion and impurities.

  It can be concluded that the carbonaceous and stainless steel lining is in satisfactory condition for continued operation.
- Condition of the hermetic steel lining of the transport corridor

  The inspection of the steel lining of the transport corridor demonstrated that the lining is intact and satisfactory for continued operation.

  The thickness of steel lining of the transport corridor is within the tolerance required by the production standard for metallurgical products and is, as a whole, satisfactory.
- Condition of the hermetic closures of containment

  The inspection of hermetic closures and doors showed that they are fully functional for the period of continued operation.
- Condition of the hermetic closures of transport corridor

  The tightness test of airlocks chambers is evaluated as satisfactory.
- Condition of hermetic penetrations from the inside of the containment

  Inspections of penetrations in the structural part are evaluated as satisfactory.
- Condition of hermetic penetrations from the outside of the containment

  Inspections of penetrations in the structural part are evaluated as satisfactory.
- Overall tightness of the containment

  The results of the pressure tests carried out so far show compliance with the allowable amount of containment leakage with a sufficient reserve. The course over time shows a gradual increase in leaks together with a gradual decrease in the trend of increase. Changes in time are similar on both units. The pressure tests of the whole containment do not allow for the identification of the locations with increase in leaks. In terms of containment structures, the local tightness tests of structures or the inspections of structures at the boundary of the containment doe not show an increase in structural leaks or defects, which could cause increase in leaks. In terms of structures, no measures are required.
- Local tightness

  In terms of containment structures, the local tightness tests of structures or the inspections of structures at the boundary of the containment doe not show an increase in structural leaks or defects, which could cause increase in leaks.

- Condition of internals concretes

  On the basis of the time characteristic of moisture values, it can be stated that the transport of moisture in the volume of internals concretes is lower than the loss of moisture as a result of ventilation of the measured points during measurement, thus drying out the concrete around the measuring points. The inflows into internals concretes through the leaks from pools can be therefore considered to be zero or very low and the initialisation of the degradation of internals structures due to leaks from pools is not foreseen.

The evaluation of inspections and measurements on the containments of Unit 1 and Unit 2 shows satisfactory condition of the containment and compliance with all design basis requirements, thus ensuring safety function of the containment.

- *AMP for Civil Structures Parts of Pools with Double Liner in the Temelín NPP*
  - Visual inspection of austenitic pool liners

    The results of the visual inspection of austenitic pool liner confirm that the surface is intact without mechanical damage and cracks. There is no deposition of boric acid salts due to solution discharges from the intermediate space into the pool, and there no blooms and mechanical deposits.

    The measured thickness of austenitic steel liner ranges from 95% to 100% of the design thickness.
  - Monitoring of pool filling cycles or water level changes

    The monitoring of pool filling/emptying cycles or water level changes shows that the measured values are below the specified limit values.
  - Temperature monitoring of permanently flooded areas over time

    The rate of change in temperature does not affect the amount of stress in liners; the magnitude of the change in temperature has a dominant influence. The measurements for the previous periods do not show any significant changes in temperature. The maximum pool temperature does not exceed or approach the limit value.
  - Detection and monitoring of seepage through the liner of all pools outside the boron pool into the intermediate space of liners

    Leaks are insignificant on a long-term basis and lining of the pools with double lining fulfils its function
  - Detection and monitoring of seepage through the liner of boron pool into the intermediate space of liners

    There is no seepage
  - Data concerning the location of spent fuel rods over time and spent fuel storage pool in relation to thermal stress of liners

    An effect of the location of spent fuel rods on stress in liners has not yet been observed

The pools with double lining in the Temelín NPP Unit 1 and Unit 2 can be considered operable on a long-term basis. The structure meets the assumptions for safe operation of the NPP in the next period.

7.2.2 LVR-15 operator's experience of the impemantation of the ageing management program for civil structures

In terms of the amount of structural response, seismic load has a dominant influence, the effects of which exceed the effects of other external events. The HCLPF (High Confidence of Low Probability of Failure) values of boundary seismic resistance calculated for component structures show that the building as a whole does not fail in a given earthquake; however, there can be partial failures of individual load bearing elements of their construction. Significant of those are cracks in backfill of the structure of reactor building. However, these cracks as a result of co-acting steel or reinforced concrete main load bearing structure do not affect robustness of the building as a whole however they can cause partial damage. It is also necessary to take breakage of window glass, glass liners and sticking of window frames, doors and gates in an earthquake into account. In order to evaluate the seismic resistance of technological equipment, the seismic response spectra in selected locations inside the buildings were also calculated.

In regards of the intensity of effects, aircraft crash and shock wave of an external explosion are other significant loads. In any case, even their effects do not cause an overall failure of buildings but only local damage such as cracks in backfill, breakage of window glass and glass liners which will be probably thrown inside the building as a result of the remaining pressure.

During an aircraft's fall down on the reactor building, especially in glass structure and external cladding, it will be broken through and a motor component of the small aircraft considered will enter the inside of the building. Breaking the aircraft in this breach will cause that the wreckage from the aircraft will have relatively small energy, which cannot cause damage to the reactor that is placed in a concrete protective shaft approximately in the centre of the reactor hall.

Additional analyses of the existing structures demonstrated that the load-bearing structures of the reactor building and the associated structures are in good condition and are able to withstand all external extraordinary influences according to existing Czech legislation and the IAEA recommendations. The assessment carried out is in full compliance with the international accepted practices for assessing research reactors.

A crack has been identified in reactor hall wall in the building in the past, which is located in the crane cab room on the third floor. The stability of this crack is monitored on a long-term basis and verified by means of a plaster cast. In 2016, this plaster cast was modified with penetration to the depth of wall core as recommended by structural engineer.

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

7.3.1 Regulator's assessment and conclusions on the ageing management of of the Dukovany and Temelín Nuclear Power Plants concrete containment structures

The SÚJB reviewed information concerning ageing management of component parts, structures and buildings that was provided for the purposes of this report by the operator of the Dukovany NPP and the Temelín NPP, together with information obtained from its inspection and assessment activities.

The state of the component parts and structures is periodically evaluated by the SÚJB. This is done under the review of information set out in the Final Safety Analysis Report updated once a year, which contains information obtained from the periodic ageing review of components parts and structures and the Ageing Management Programmes for components parts of both NPPs as well as during periodic planned or random inspection activity.

In their inspection and assessment activities, inspectors verify and evaluate information concerning the component parts and structures, plant's ability to perform its functions and evaluate other documentation demonstrating the ability of component parts and structures to perform their functions (in particular, to prevent the release of radioactive material and ionising radiation into the environment – strength and tightness functions of the containment system). Furthermore, compliance of the activities carried out by the holder of licese under specific processes with the relevant requirements of existing legislation and normative documents concerning the component parts and structures is controlled and assessed.

During the licensing process for the operation of Dukovany NPP units after 30 years of operation (i.e. for "LTO"), the whole ageing management system for structures was examined in detail. With a comprehensive assessment of the issue of structures, the SÚJB identified, on the basis of feedback from NPP operation and the results of its own inspection and assessment activity, minor weaknesses inrecording and documenting of the as-built configuration of buildings and structures. Compared to the good worldwide practice, not all Ageing Management Programmes have been set and implemented, the AMP for Monitoring of Structures the AMP for Monitoring of Buildings, and recording and documenting of data related to maintenance and testing of component parts in particular. ČEZ, a. s. developed an action plan, proposing remedial actions in response to weaknesses identified, together with the dates for their implementation, committing the applicant to eliminate such weaknesses. In addition, the applicant provided, in the course of administrative procedure, additional information and data obtained from containment integrity verification test, which documented the condition and ability to perform the safety function of the containment system in DBA.

In general, all information available was evaluated by the Office as satisfactory, with several formal weaknesses, which do not prevent the safe operation of the Dukovany NPP. However, in taking the Decisions on Operation, the SÚJB found them to be valuable in terms of continuous improvement of the level of nuclear safety and the deadlines for their elimination were set under the conditions of the issued Decisions on Dukovany NPP units operation.

The Ageing Management Programmes are currently set in accordance with the requirements set out in existing legislation and the good worldwide practice for both NPPs. Under the Ageing Management Programmes, the holder of the licence to operate the Dukovany NPP or its suppliers carries out the relevant controlled activities. These include: performance, evaluation and documentation of periodic inspections against the passport of defects and failures on structures; fulfilment of the programme for monitoring of buildings; measurements of settlements and deflections of safety relevant buildings; visual inspections and measurements at selected locations for the condition of hermetic steel lining; determination and periodic verification of the tightness of containment during periodic tests; testing of hermetic lids, closures and doors at the boundary and within the hermetic area; measurements of temperature patterns; deformation measurements of the roof and the shaft wall by localising accidents; planned performance and evaluation of the Institute of Concrete Surveillance Specimens, and the implemented research programme for the boric acid influence on the concrete. In addition, under the AMP for Fuel Storage and Refuelling Pools in the Dukovany NPP, the following activities are carried out: monitoring and evaluation of the amount of the leak detected in the controlled leakage system (TZ system) from the spent fuel storage pools; monitoring and recording of filling and emptying cycles; solution temperature pattern; visual inspections of loss of integrity, damage to the surface of stainless steel liner above the water level by cracks. The last of the monitored parameters above is a new monitored parameter and will be evaluated in 2018.

In addition, the Ageing Management Programme for the Temelín NPP includes: pre-stressing force test; measurement of structural response to the load; state of the pre-stressing system and condition of the cover of dome, cornice and bearing ring. These programmes are based on the different design of the containment for both NPPs.

The SALTO mission, in the assessment of Dukovany NPP preparedness for long-term operation, found minor weaknesses, which were related to the identification of relevant degradation mechanisms/ageing effects and incorporation of data from walkdowns, including operating experience of the Dukovany NPP and international experience into the Annual Final Assessment Report for each civil structure being assessed. The holder of the license eliminated such weaknesses within the proposed time limits by having a passportization of civil structures made by suppliers and then supplemented the final report on structures and incorporated it into the recent AMP Monitoring of Buildings. Its copy is regularly updated in the Final Safety Analysis Report.

The long-term bottleneck is the question of monitoring the condition of stainless steel liner of spent nuclear fuel storage pools, where access as well as means for identifying the state and extent of the effects of degradation mechanisms on primary stainless steel and secondary steel liner of storage pools in operation or in filled (full) state, are restricted. The holder of the licese addressed this issue through an investment project, which includes an analysis of options to achieve improvement in this field.

Despite the above mentioned problematic topics on the monitoring of the condition of hermetic liner of storage pools, the SÚJB evaluated the Ageing Management Programmes for component parts and structures for the Dukovany NPP and the Temelín NPP as adequately set and sufficiently effective, and in its inspection and assessment activity the SÚJB takes care to ensure their compliance and periodic review by the holder of license, and highlights their improvement, applying the latest knowledge of science and technology, and good worldwide practice.

7.3.2 Regulator’s assessment and conclusions on the ageing management of the LVR-15 civil structures

The structures of the LVR-15 research reactor are currently not included within the scope of the Ageing Management Programme for the LVR-15 Reactor. The civil structures are monitored in the framework of period inspections under the In-service Inspection Programme. Furthermore, in response to events at the Fukushima Daiichi NPP, structural robustness to withstand external influences was assessed. In view of the new legislation issued, the Ageing Management Programme for the LVR-15 reactor will be adapted to the new Atomic Act by the end of 2018 (transitional provisions). Compliance with the requirements set out in new legislation (including incorporation of important civil structures) will be then reviewed by the SÚJB.

8. Pre-stressed concrete pressure vessels (AGR)

No AGR type reactors are in operation in the Czech Republic.

9. Overall assessment and general conclusions

Generic requirements on ageing management were included in the legislation documents since nuclear utilisation in Czech Republic beginning i.e. in Act No. 28/1984 Coll. and its implementing decrees. All this documents were updated and upgraded according to research and development results, operational experience and the growing needs for nuclear safety improvement.

The detailed specific requirements for ageing management are implemented in national legislation of the Czech Republic. On 1 January 2017, a new Atomic Act came into force, which includes the requirements for implementation of the ageing management process defined in the Ageing Management Programmes, with detailed specification of these requirements in the implementing regulations. The requirements arising from the IAEA Safety Principles and Requirements and WENRA Safety Reference Levels (Criteria) are incorporated into new legislation.

Monitoring of reactor pressure vessels ageing was carried out from the beginning of operation of both nuclear power plants. Complex monitoring and evaluation of residual lifetime of the main primary circuit components and other safety-critical SSCs were introduced gradually.

The previous legislation, in force till the end of 2016, included the requirement for the identification of the criteria for life monitoring included in the final output documentation for the process of designing the selected equipment included in Safety Class 1 or 2 and other implicit obligations. The definitions and procedures for systematic approach on ageing management are the subject of Regulatory Safety Guide BN-JB-2.1 "Ageing Management for Nuclear Installations" [5]. SÚJB intends to revise this guide in the light of new legislative framework. Furthermore, the requirement for ageing monitoring and prediction of the residual life of the most important components in a comprehensive and systematic way was included in the Decisions on Operation of the Dukovany NPP Units after 10 and 20 years of operation as well as in the Decisions on Operation of the Temelín NPP Units. The results of such reviews are updated on a yearly basis and transferred into the Final Safety Analysis Report. For nuclear research reactors, just a brief guidance for ageing monitoring of such facilities is defined in the legislation.

Pursuant to new Atomic Act, the holder of the licence to operate the NPP (ČEZ, a. s.) has the essential requirements for ageing management implemented in its internal processes. The implementation of the requirements for major components or relevant degradation mechanisms is described in the set of Component Specific Ageing Management Programmes or Specific Ageing Management Programmes. This set is updated on the basis of periodic assessment of the effectiveness of ageing management (with the use of internal and external feedback from operation, current science and technology state-of-the-art, a research results etc.). The process under which ageing management is carried out and its implementation for the components included within the scope of equipment to demonstrate reliability in operation beyond the design limit were verified in the licensing procedure process for the operation of Dukovany NPP after 30 years of operation. The requirements for addressing the identified weaknesses were specified in the Decisions on Operation Dukovany NPP units after 30 years of operation. Furthermore, the results of periodic safety review were assessed by the State Office for Nuclear Safety, and ageing management process at the level of individual components or degradation mechanisms is also monitored in the framework of inspection activity.

Proving validity and sufficiency of safety margins as well as an ageing criteria used in the AM process is the beginning of the AMR. During ageing program assessments the holder of license as well as regulator found out the need to verify the design documentation and its design basis. Also the contractors´ manufacturing and maintenance documentation was reviewed and completed.

Due to different reasons equipment configuration is going through changes during the NPP life. Ageing is one of many time dependent processes, those monitoring and assessment is sophisticated engineering discipline. That's why configuration management process is considered in Czech Republic as a key supporting process of ERM.

From an international point of view, the system was verified by full-scope SALTO missions that took place at the Dukovany NPP in 2008 (follow-up in 2011) and 2014 (follow-up in 2016). ČEZ, a.s. is also an active member of IAEA IGALL project whose results are subsequently implemented in the ageing management programmes.

The current state and lifetime prediction of cables is periodically assessed by SÚJB inspectors under the review of the Final Safety Analysis Report which is updated on an annual basis. The Final Safety Analysis Report provides information from annual life assessment of the cables within the scope of the Ageing Management Programme for Cables (AMPC). SÚJB inspectors periodically assess the condition of cable sets, and the activities carried out during and outside outages are assessed and controlled (inspections, replacements, reconstructions, etc.) within their inspection and assessment activities. The Programme is set for all safety-related cables, regardless of whether they are high-voltage or low-voltage cables. A whole range of activities is carried out under the Ageing Management Programme for Cables, from cable qualification for the harsh environment, monitoring and evaluation of environmental parameters at locations where cables are installed, visual inspections of installed cables, assessment of the cables removed during technology restoration, installation of cables in deposits (the surveillance programme). In the operating experience feedback, there were any serius problems relating to cable sets. The AMP for Cables has been recognised at the international level – by SALTO mission assessing Dukovany NPP preparedness for long-term operation as well as by the EPRI (the AMPC won the award for the implementation of the Ageing Management Programme for cables in 2016). This could be considered as a "Good performance" and/or "Good practice".

In summary, the AMPC is implemented on such a level allowing sufficient prediction of changes in a timely manner.

The SÚJB throughout the condition of the Decision on Dukovany NPP Operation imposed an obligation on the operator to put into practice the methodology for monitoring physical condition of the ESW system (including inaccessible pipework). The frequency and scope of the inspections had to be set so as to reveal, sufficiently in advance, any irregularities and defects caused by system operation, thus preventing significant malfunctions of that system.

On that basis the In-Service Inspection Programme of ESW system was improved and the Ageing Management Programme for Concealed Pipework and the Ageing Management Programme for Service Waters were implemented. Implementation of these programmes was preceded by research projects or activities, e.g. development of the EDMET method, cooperation with the EPRI in the implementation of the BPWORKS™ application. From the perspective of the SÚJB, both Programmes formally meet the attributes required by legislation of the Czech Republic; nevertheless, due to the recent date of their implementation, conclusions cannot be draw yet on their effectiveness. SÚJB monitors the activities carried out under that programme in the framework of its assessment and inspection activities.

The Component specific AMP for Reactor (RPV is a part of this particular AMP) is based on a broad spectre of activities. The most important ones are monitoring and evaluation of materials irradiation embrittlement and results of qualified In-service Inspections of welds, cladding and base metal of RPV. To the final conclusion on residual lifetime and lifetime prediction enter also results of fatigue evaluation, thermal ageing etc. he activities focused on the monitoring of current condition and lifetime assessment of reactor pressure vessel were carried out from the beginning of operation and were expanded on the basis of the current state of knowledge and operation feedback. Changes were addressed to the optimisation of fuel loading pattern (low-leakage zone), development of Surveillance Specimen Programme from the Standard one to the Supplementary Surveillance Programme and subsequently to the Extended Surveillance Programme to cover LTO phase of Dukovany NPP. Also the changes (reduction) in heat-up and cool down rate to reduce the fatigue stress or changes to In-service Programme were performed.

The current Component Specific Ageing Management Programme for Reactor covers all significant and expected degradation mechanisms and is set in accordance with the international best practices. The results of the periodic life assessment of reactor are presented in the Final Safety Analysis Report updated once a year and reviewed by the SÚJB. In the framework of inspection activity, particular attention is paid to the current results of in-service inspections and maintenance practices, any modifications and repairs during refuelling outages on individual units.

The Ageing Management Programme for Reactor meets the requirements set out in applicable legislation and other documents within the scope of national legislative and regulatory framework. SÚJB considers the Component Specific Ageing Management Programme for Reactors of the Dukovany and Temelín NPPs to be properly set and implemented and sufficiently effective.

Ageing management process of component parts, structures and buildings was completed recently. Incompleteness of AMPs for civil structures was also finding from the IAEA SALTO mission on Dukovany NPP preparedness for LTO. On the basis of SÚJB review, evaluation and inspection work, the requirements on improvement of the AM process for civil structures were formulated, i.e:

- Holder of license should prepare civil structures AM documentation (including AMR, Health Reports, TLAA and final report of Effective maintenance strategy programme) addition of characteristic and all ageing degradation mechanisms results with impact to safety functions performance.
- Holder of license should include in FSAR long term monitoring results of safety important civil structures subsidence, pasportisation of defects and faults of civil SSC as well as results of civil SSC ageing assessments.
- Holder of license should implement Ageing Management program “Monitoring of civil SSC” for civil SSC important to nuclear safety and information will be submitted to SÚJB via the latest revision of FSAR.
- Holder of license should develop and implement methods for hermetic liner monitoring which will be part of In-service Inspection Program.

At the moment the above mentioned requirements that were formulated as Conditions of Decision on Dukovany NPP Operation are fulfilled and the required actions are implemented. The ageing management activities comply with the international good practice.

Last AMR and related licensing process performed at Dukovany NPP at the end of its designed life was very thorough activity. Regardless of the good internal or external appraisals mentioned above, this process revealed the need for improvement. A lot of compensatory work has been done in time, but in some areas, which are not inevitable for further operation, but improvements of some processes and documents is needed. This was recognized by SÚJB in it's review and inspection activities and resulted in number of conditions in the Decisions on Operation of Dukovany NPP Units. Examples of these conditions related to ageing management area are (in simplified wording) as follow:

- Holder of license shall once a year submit to SÚJB summary of actualized Final Safety Analysis Report which must reflect actual status of EDU and provide summary information on condition of selected equipment and its residual age.
- Holder of license shall prepare and submit to SÚJB the validity assessment of PIE analysis (break of high and middle energetic pipelines and fatigue damage of hermetic liners) and update the TLAA database.
- Holder of license in relation to AMR and In-service inspection program shall verify sensitivity of steam-generator RT inspections method and based on the results to propose the program modification.
- Holder of license shall keep constantly up-to-date the documentation on AM status and conditions of selected equipment, civil structures important to assure the safety functions and equipment which, if failed, can endanger selected equipment functionality (i.e. AMR, HR, TLAA and maintenance documents)

For the LVR-15 nuclear research reactor, the Ageing Management Programme was developed, assessing the relevant components in terms of ageing effect and their life prediction. On the basis of the analyses, remedial actions were taken to ensure the safe operation for at least 10 years beyond the design limit. The requirements set out in new legislation are not fully implemented with the holder of the licence to operate this nuclear facility; fulfilment of these requirements will be reviewed by the State Office for Nuclear Safety following the transitional provisions of the new Atomic Act.

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[111] Regulations for testing welded joints and cladding of flanges and structures of nuclear power plants and nuclear research reactors and facilities PK1514/72, Gosgortechnadzor, 1974

[112] Guidelines and recommendations regarding ageing assessment of NPP VVER reactor pressure vessel and internals during NPP operation, issued by the State Office for Nuclear Safety in 12/1998

[113] Prediction of Mechanical Properties of Irradiated Austenitic Stainless Steels, ENES, Moscow 2007

[114] ČEZ_ME_1029 AMP for Monitoring of Dukovany NPP Buildings

[115] ČEZ_ME_1030 AMP for Monitoring of Structures of the Dukovany NPP

[116] ČEZ_ME_0934 AMP for Measuring of Civil Structure Settlement

[117] ČEZ_ME_0937 AMP for Containments in the Dukovany NPP

[118] ČEZ_ME_0936 AMP for Fuel Storage and Refuelling Pools in the Dukovany NPP

[119] ČEZ_ME_0966 AMP for Civil Structure Parts of Containment in the Temelín NPP

[120] ČEZ_ME_0964 AMP for Civil Structure Parts of Pools with Double Liner in the Temelín NPP

[121] IAEA NP-T-3.5, Ageing Management of Concrete Structures in Nuclear Power Plants, Vienna, 2016

[122] ACI 349.3R-02, Evaluation of Existing Nuclear Safety-Related Concrete Structures, Ronald J. Janowiak a spol, 2002

[123] ČSN ISO 13822 Bases for design of structures - Assessment of existing structures

[124] INPO AP-913, Equipment Reliability Process Description, Rev. 4, 2013

Annex A: Figures

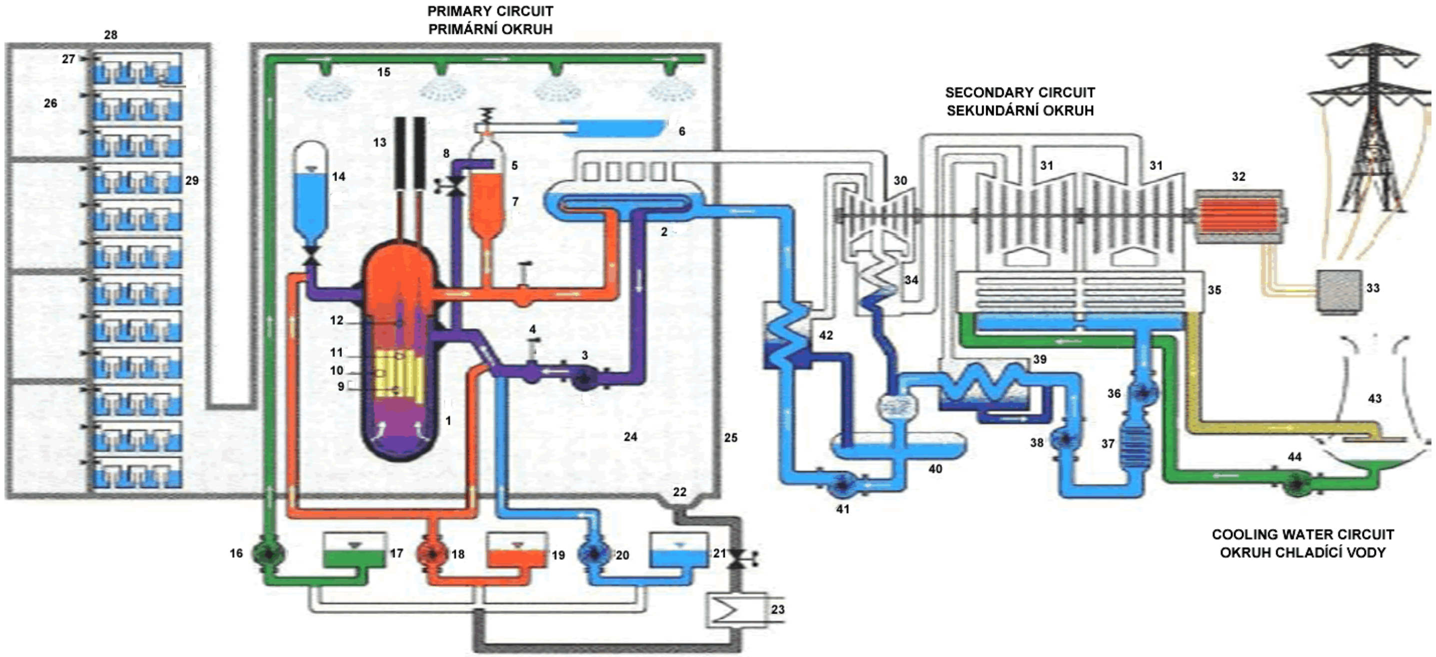


Fig. A.1: General layout of Dukovany NPP

Legend:

1 – reactor, 2 – steam generator, 3 – main circulation pump, 4 – main isolation valve, 5 - pressurizer – steam, 6 – bubbler tank, 7 – KO – water, 8- KO injections, 9 – core, 10 – fuel assembly, 11 – control rod, follower, 12 – control rod , absorber, 13 – control rod drives, 14 – hydroaccumulator, 15 – spray system, 16 – spray pump, 17 – spray system storage tank, 18 – low-pressure safety injection pump, 19 – storage tank of the low-pressure safety injection system, 20 – high-pressure safety injection pump, 21 - storage tank of the high-pressure safety injection system, 22 – suction from the hermetic zone, 23 – spray system cooler, 24 – containment, 25 – primary containment, 26 – gas holders of the bubbler tower, 27 – check flap valve, 28 – bubbler tower, 29 – bubbler tower´s conduits, 30 – high-pressure part of the turbine, 31 – lowpressure part of the turbine, 32 – electric generator, 33 – generator transformer, 34 – separator and steam reheater, 35 – condenser, 36 – condensate pump I*, 37 – condensate demineraliser, 38 – condensate pump II*, 39 – low-pressure regeneration, 40 – feed water tank, 41 – auxiliary feed water pump, 42 – high-pressure regeneration, 43 – cooling tower for circulation water, 44 – CV pumps

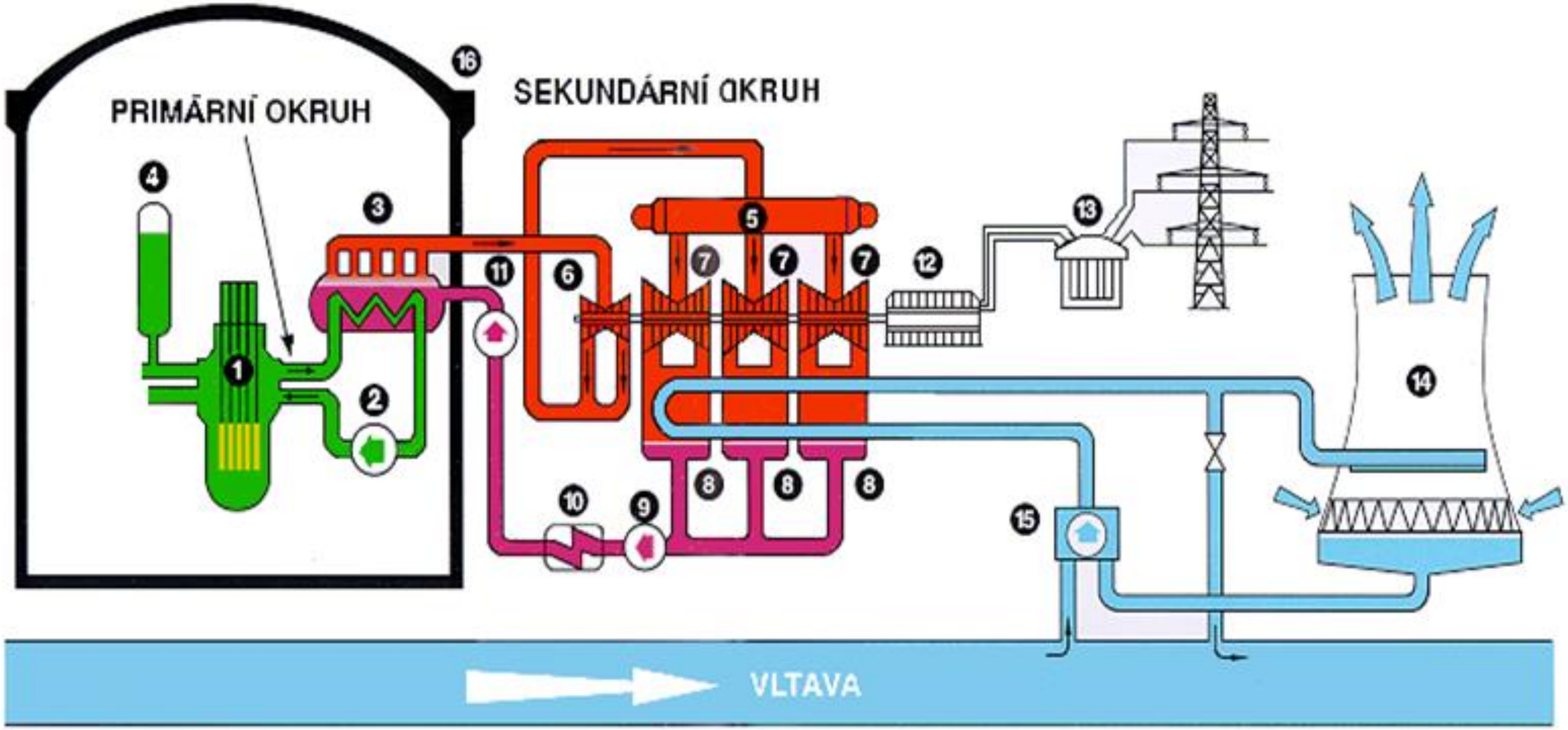


Fig. A.2: General layout of Temelín NPP

Legend: 1 – reactor, 2 – main circulation pumps, 3 – steam generator, 4 – pressurizer, 5 – separator – reheater, 6 – HP turbine, 7 – LP turbine, 8 – condenser, 9 – condensate pump, 10 – regeneration, 11 – feedwater pump, 12 – generator, 13 – transformer, 14 – cooling tower, 15 – circulation cooling water pumping station, 16 – containment

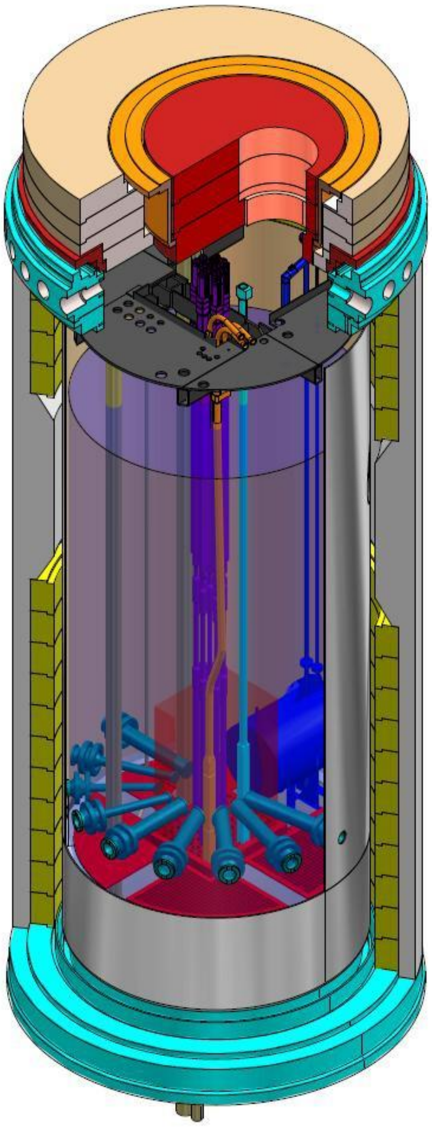
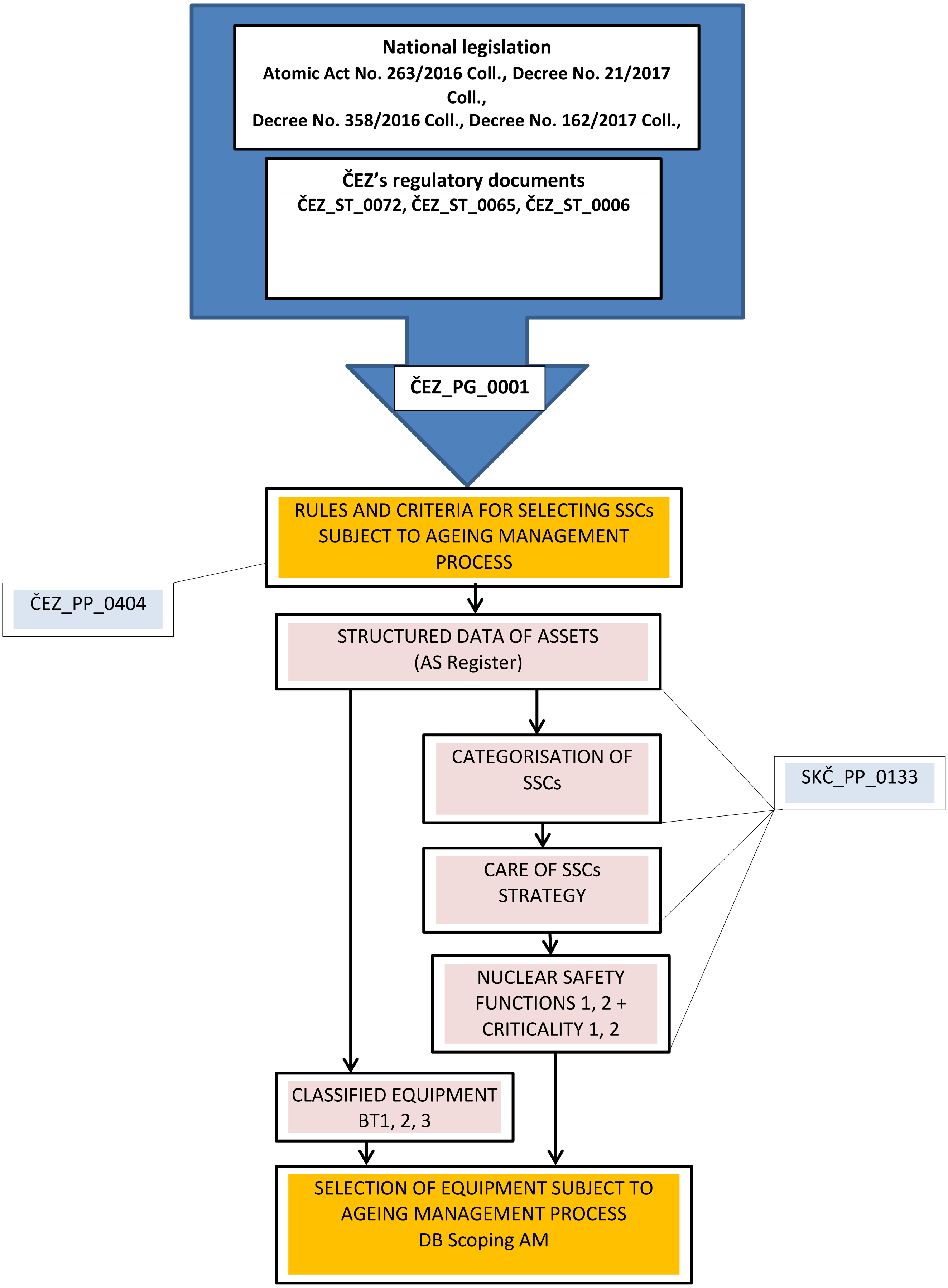
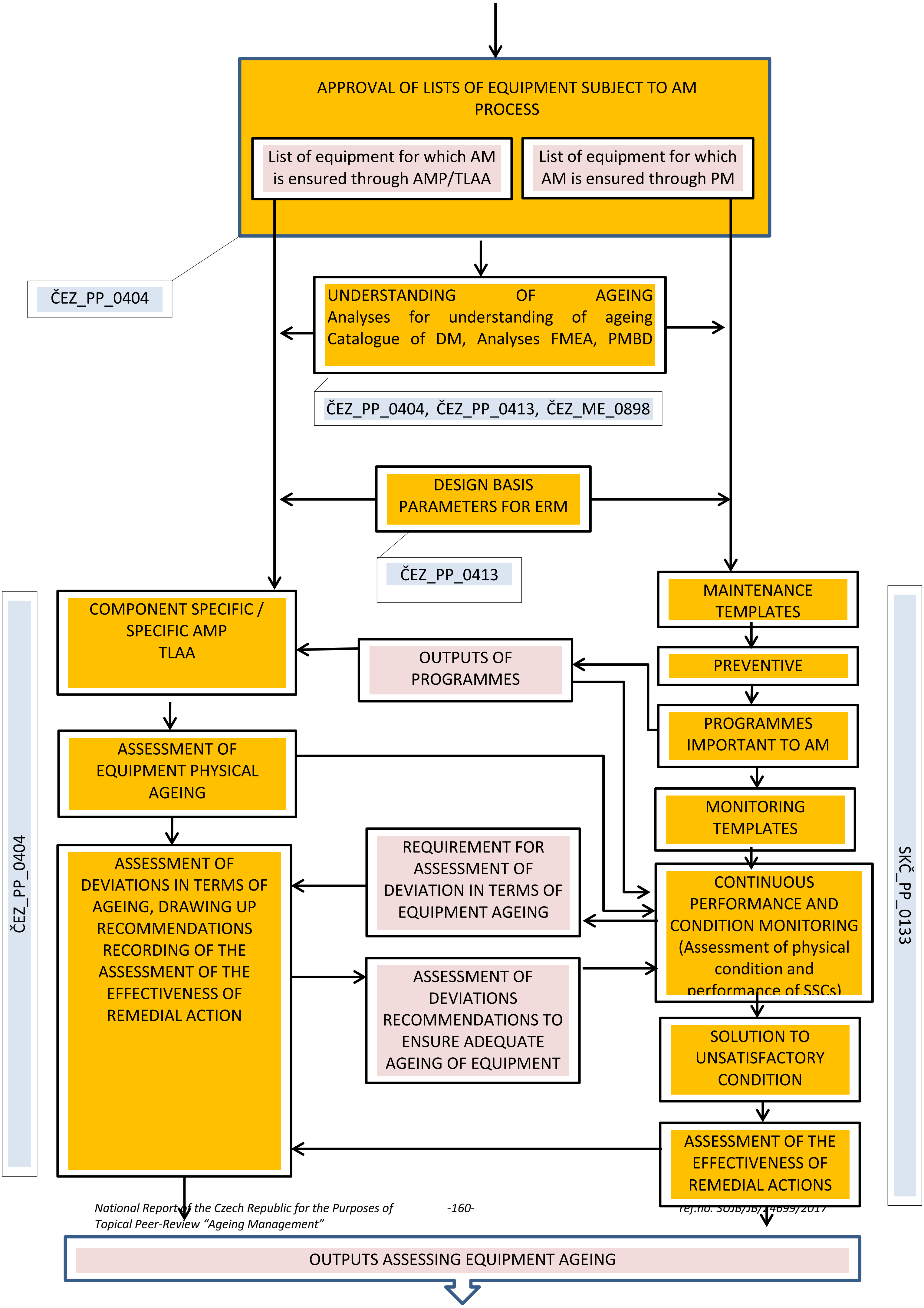


Fig. A.3: General layout of LVR-15 research reactor


National legislation
Atomic Act No. 263/2016 Coll., Decree No. 21/2017 Coll.,
Decree No. 358/2016 Coll., Decree No. 162/2017 Coll.,
ČEZ's regulatory documents
ČEZ_ST_0072, ČEZ_ST_0065, ČEZ_ST_0006
ČEZ_PG_0001
RULES AND CRITERIA FOR SELECTING SSCs SUBJECT TO AGEING MANAGEMENT PROCESS
ČEZ_PP_0404
STRUCTURED DATA OF ASSETS (AS Register)
CATEGORISATION OF SSCs
SKČ_PP_0133
CARE OF SSCs STRATEGY
NUCLEAR SAFETY FUNCTIONS 1, 2 + CRITICALITY 1, 2
CLASSIFIED EQUIPMENT BT1, 2, 3
SELECTION OF EQUIPMENT SUBJECT TO AGEING MANAGEMENT PROCESS
DB Scoping AM



APPROVAL OF LISTS OF EQUIPMENT SUBJECT TO AM PROCESS
List of equipment for which AM is ensured through AMP/TLAA
List of equipment for which AM is ensured through PM
ČEZ_PP_0404
UNDERSTANDING OF AGEING Analyses for understanding of ageing Catalogue of DM, Analyses FMEA, PMBD
ČEZ_PP_0404, ČEZ_PP_0413, ČEZ_ME_0898
DESIGN BASIS PARAMETERS FOR ERM
ČEZ_PP_0413
MAINTENANCE TEMPLATES
PREVENTIVE
PROGRAMMES IMPORTANT TO AM
MONITORING TEMPLATES
CONTINUOUS PERFORMANCE AND CONDITION MONITORING (Assessment of physical condition and performance of SSCs)
SOLUTION TO UNSATISFACTORY CONDITION
ASSESSMENT OF THE EFFECTIVENESS OF REMEDIAL ACTIONS
SKČ_PP_0133
COMPONENT SPECIFIC / SPECIFIC AMP TLAA
OUTPUTS OF PROGRAMMES
ASSESSMENT OF EQUIPMENT PHYSICAL AGEING
ASSESSMENT OF DEVIATIONS IN TERMS OF AGEING, DRAWING UP RECOMMENDATIONS RECORDING OF THE ASSESSMENT OF THE EFFECTIVENESS OF REMEDIAL ACTION
REQUIREMENT FOR ASSESSMENT OF DEVIATION IN TERMS OF EQUIPMENT AGEING
ASSESSMENT OF DEVIATIONS RECOMMENDATIONS TO ENSURE ADEQUATE AGEING OF EQUIPMENT
ČEZ_PP_0404
OUTPUTS ASSESSING EQUIPMENT AGEING

I. Ageing assessment of safety relevant equipment Category A *(period of update: 1 year)*

II. Assessment of validity conditions for TLAA*(period of update: 1 year)*

III. Chapter 13.4.8 Safety Report *(period of update: 1 year)*

IV. Chapter 7 Health Report *(period of update: 1 year)*

V. Annual Ageing Assessment Report for Dukovany NPP/Temelín NPP *(separate reports for Dukovany NPP, Temelín NPP)*

VI. Review of ageing management - "partial AMRs" *(period of update: 5 years)*

VII. Feasibility study *(Technical part of the Technical-economic Study, deadline according to the LTO project)*

VIII. Certificate of the readiness of equipment for LTO *(deadline according to the LTO Programme)*

IX. Outputs of V01.03, i.e. output of the "Continuous performance and condition monitoring".

Fig. A.4 Scheme of the Ageing Management for NPPs

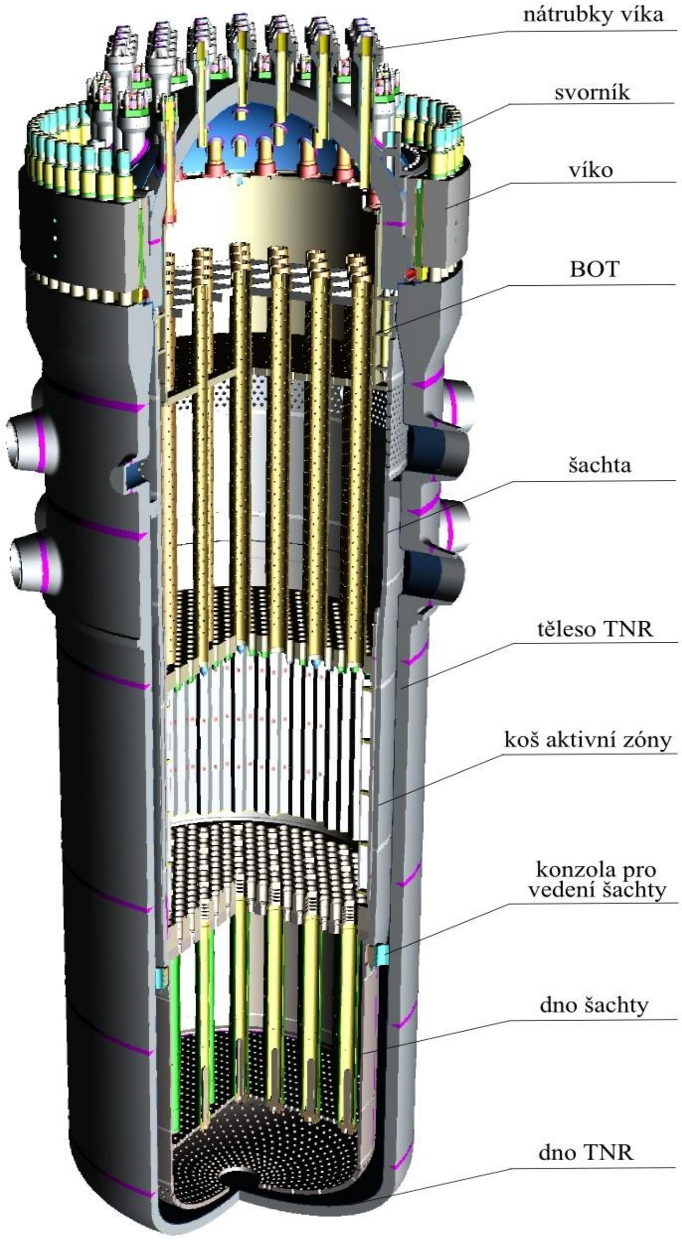


Fig. A.5 VVER 440/213-Č reactor

Legend

Nátrubky víka	Head nozzles
Svorník	Stud
Víko	Vessel head
BOT	Protective tubes block
Šachta	Reactor cavity
Těleso TNR	RPV body
Koš aktivní zóny	Core basket
Konzola pro vedení šachty	Bracket for cavity guide
Dno šachty	Cavity bottom
Dno TNR	RPV bottom

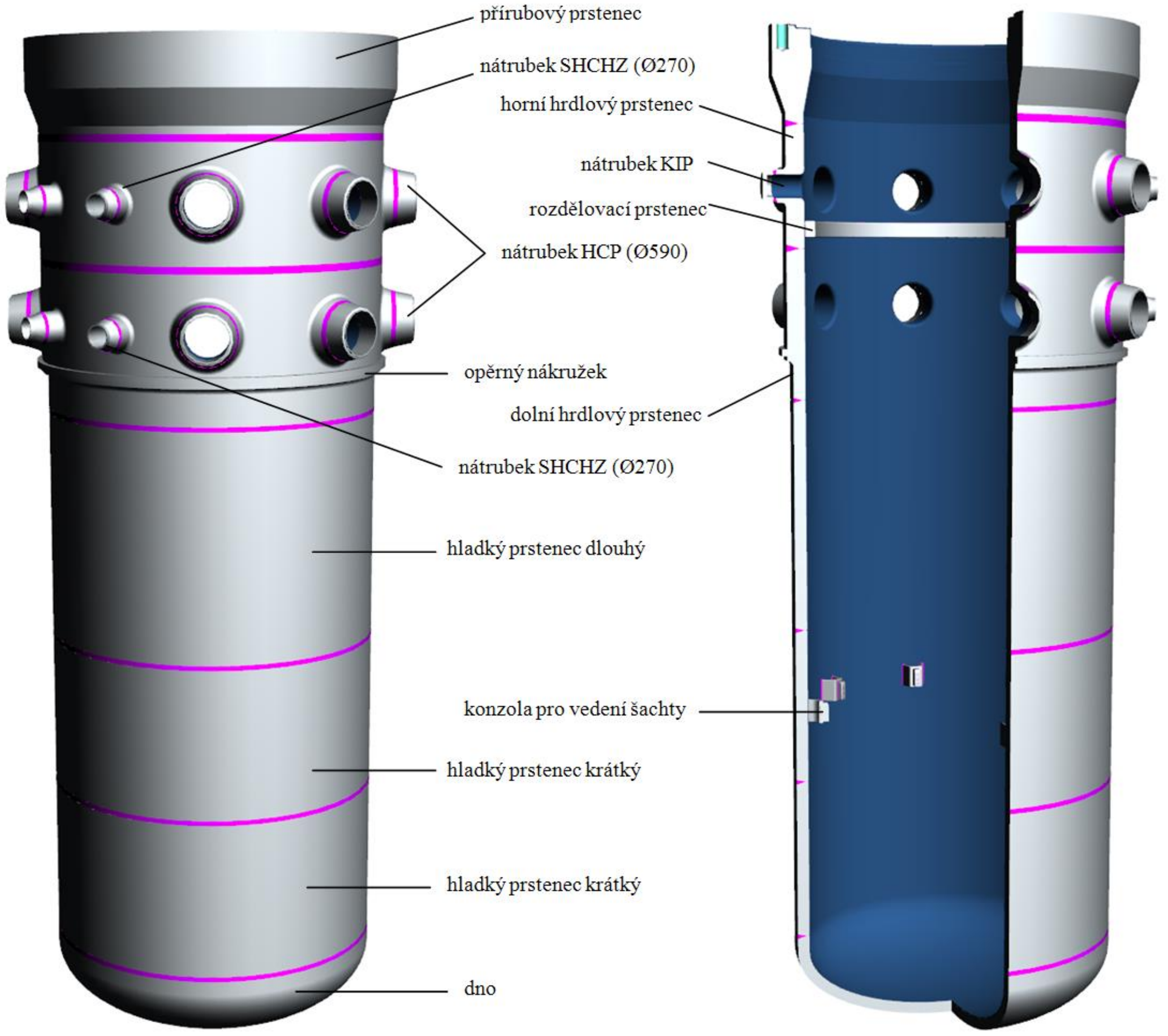


Fig. A.6: Reactor pressure vessel body

Legend

Přírubový prstenec	Flange ring
Nátrubek SHCHZ	ECCS nozzle
Horní hrdlový prstenec	Upper nozzle ring
Nátrubek KIP	I&C nozzle
Rozdělovací prstenec	Partition ring
Nátrubek HCP	Main circulation piping nozzle
Opěrný nákružek	Loose collar
Dolní hrdlový prstenec	Lower nozzle ring
Hladký prstenec dlouhý	Smooth ring long
Konzola pro vedení šachty	Bracket for cavity guide
Hladký prstenec krátký	Smooth ring short
Dno	Bottom

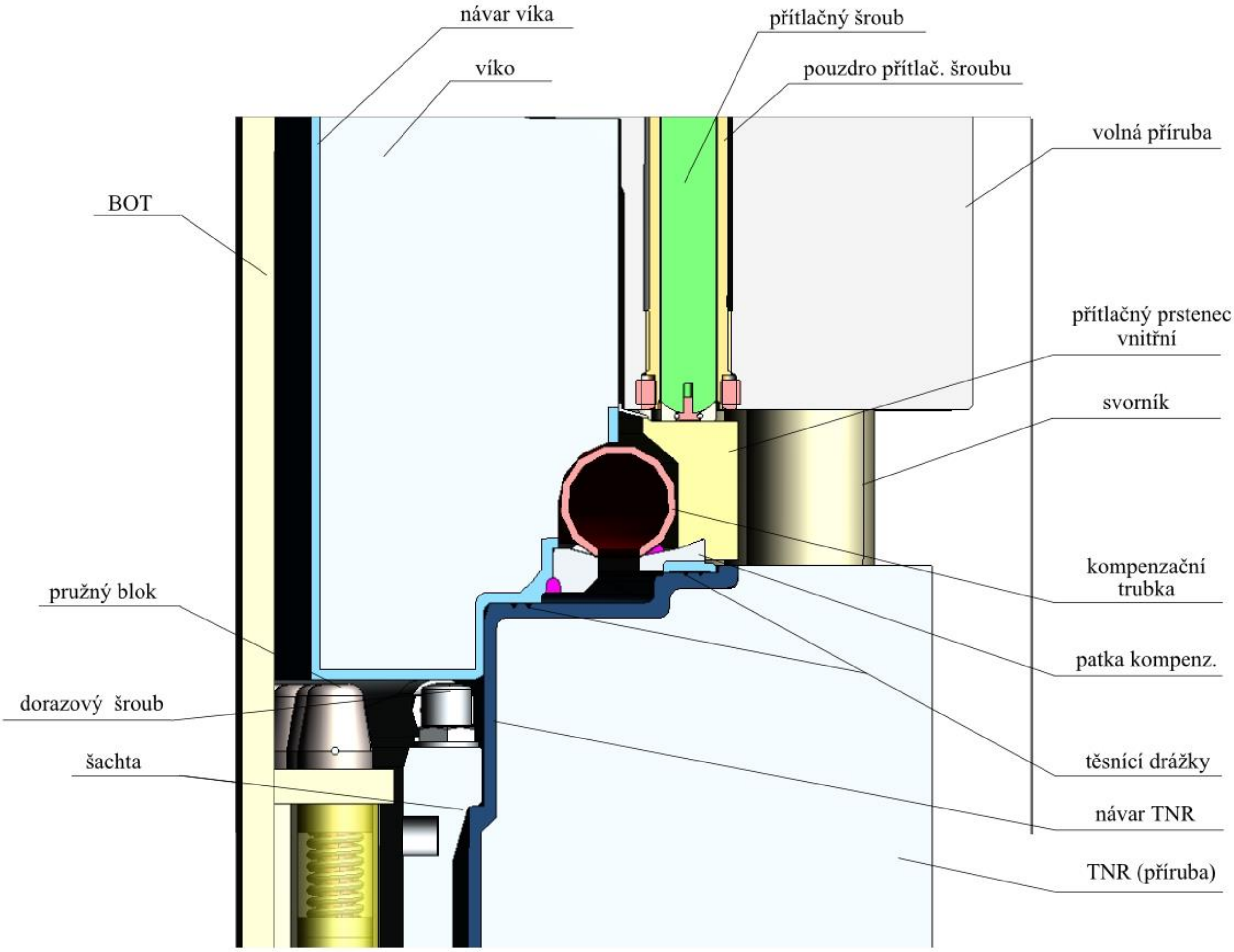


Fig. A.7: Reactor parting line detail (flange)

Legend

BOT	Protective tubes block
Pružný blok	Elastic block
Dorazový šroub	Stop screw
Šachta	Cavity
Návar víka	Head cladding
Víko	Vessel head
Přítlačný šroub	Pressure screw
Pouzdro přítlač. šroubu	Pressure screw bushing
Volná příruba	Floating flange
Přítlačný prstenec vnitřní	Pressure ring inner
Svorník	Stud
Kompenzační trubka	Expansion compensating pipe
Patka kompenz.	Compensating foot
Těsnící drážky	Sealing grooves
Návar TNR	RPV cladding
TNR (příruba)	RPV (flange)

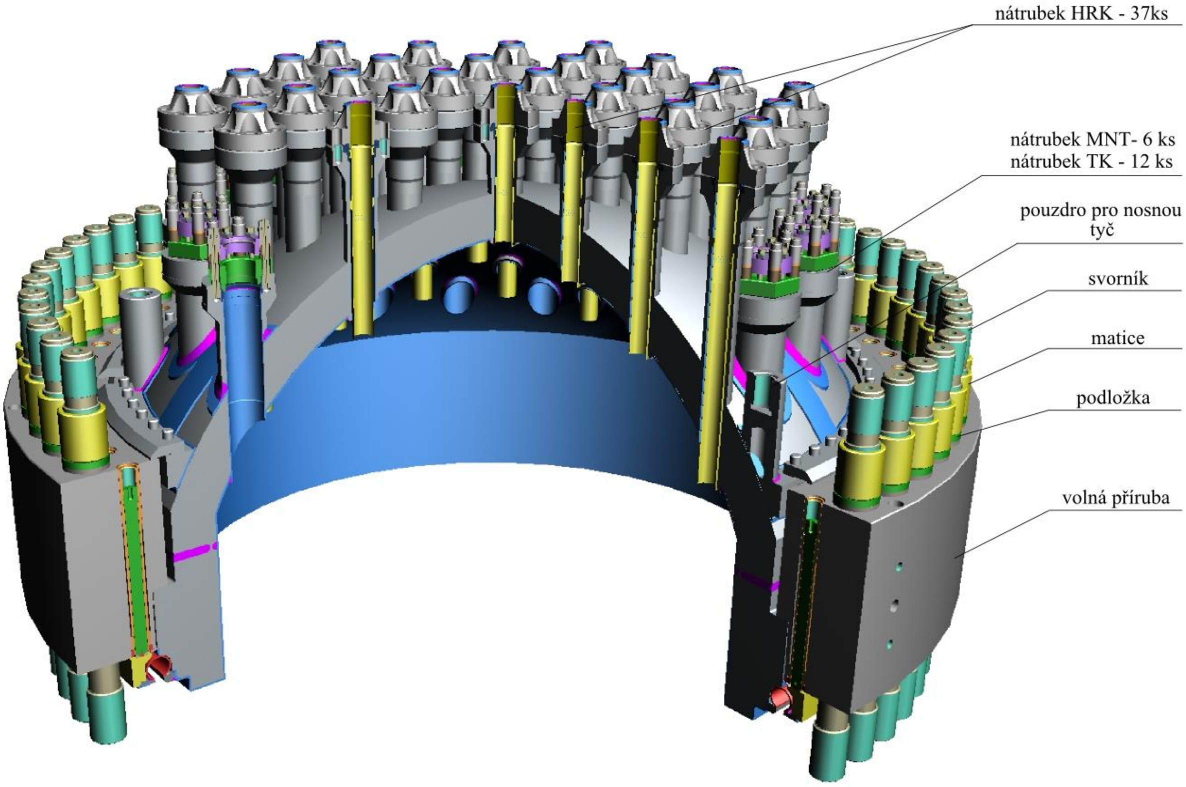


Fig. A.8: Reactor pressure vessel head

Legend

Nátrubek HRK – 37ks	CRDM nozzle – 37 pcs
Nátrubek MNT – 6 ks	NFM nozzle – 6 pcs
Nátrubek TK – 12 ks	TM nozzle – 12 pcs
Pouzdro pro nosnou tyč	Bushing for support rod
Svorník	Stud
Matice	Nut
Podložka	Washer
Volná příruba	Floating flange

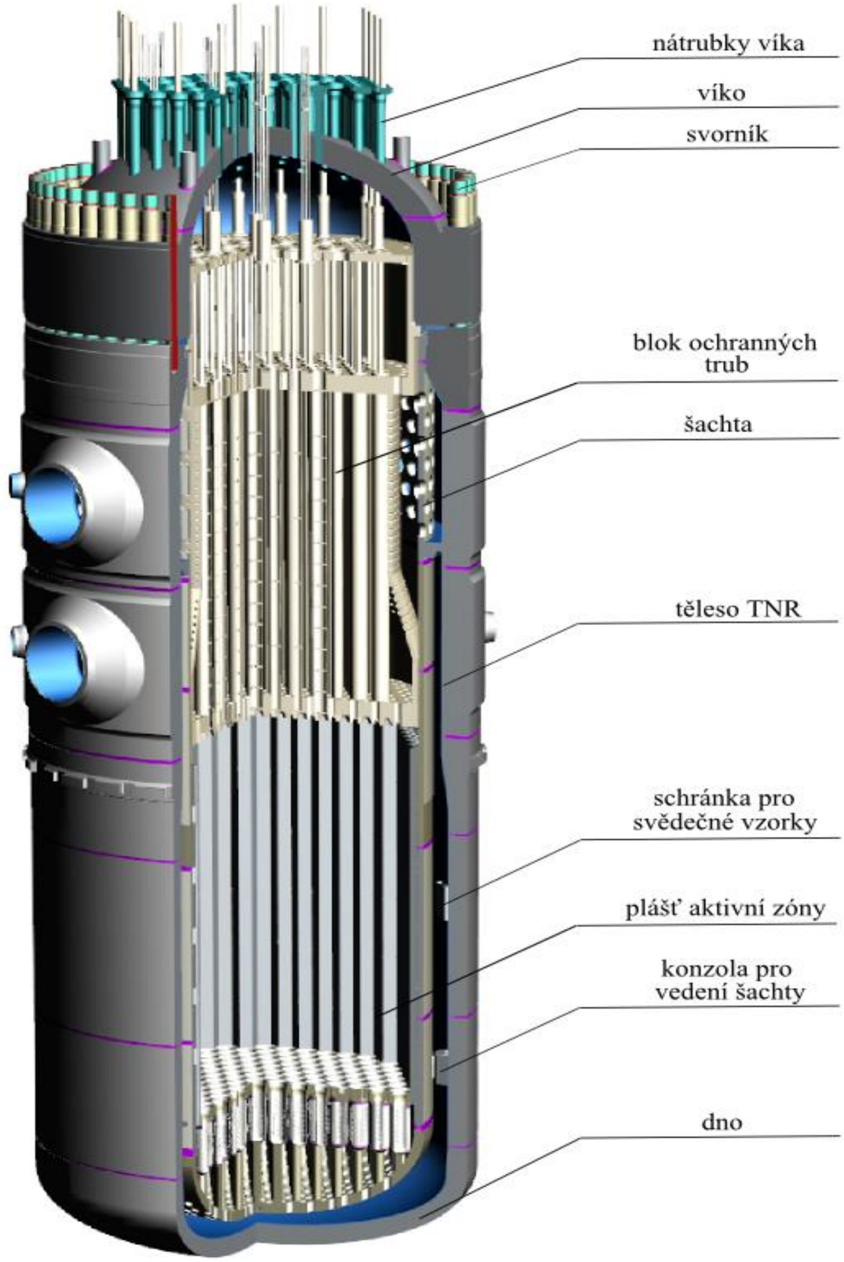


Fig. A.9: VVER 1000 reactor

Legend

Nátrubky víka	Head nozzles
Svorník	Stud
Víko	Vessel head
Blok ochranných trub	Protective tubes block
Šachta	Reactor cavity
Těleso TNR	RPV body
Schránka pro svědečné vzorky	Container for surveillance specimens
Plášť aktivní zóny	Core baffle
Konzola pro vedení šachty	Bracket for cavity guide
Dno	Bottom

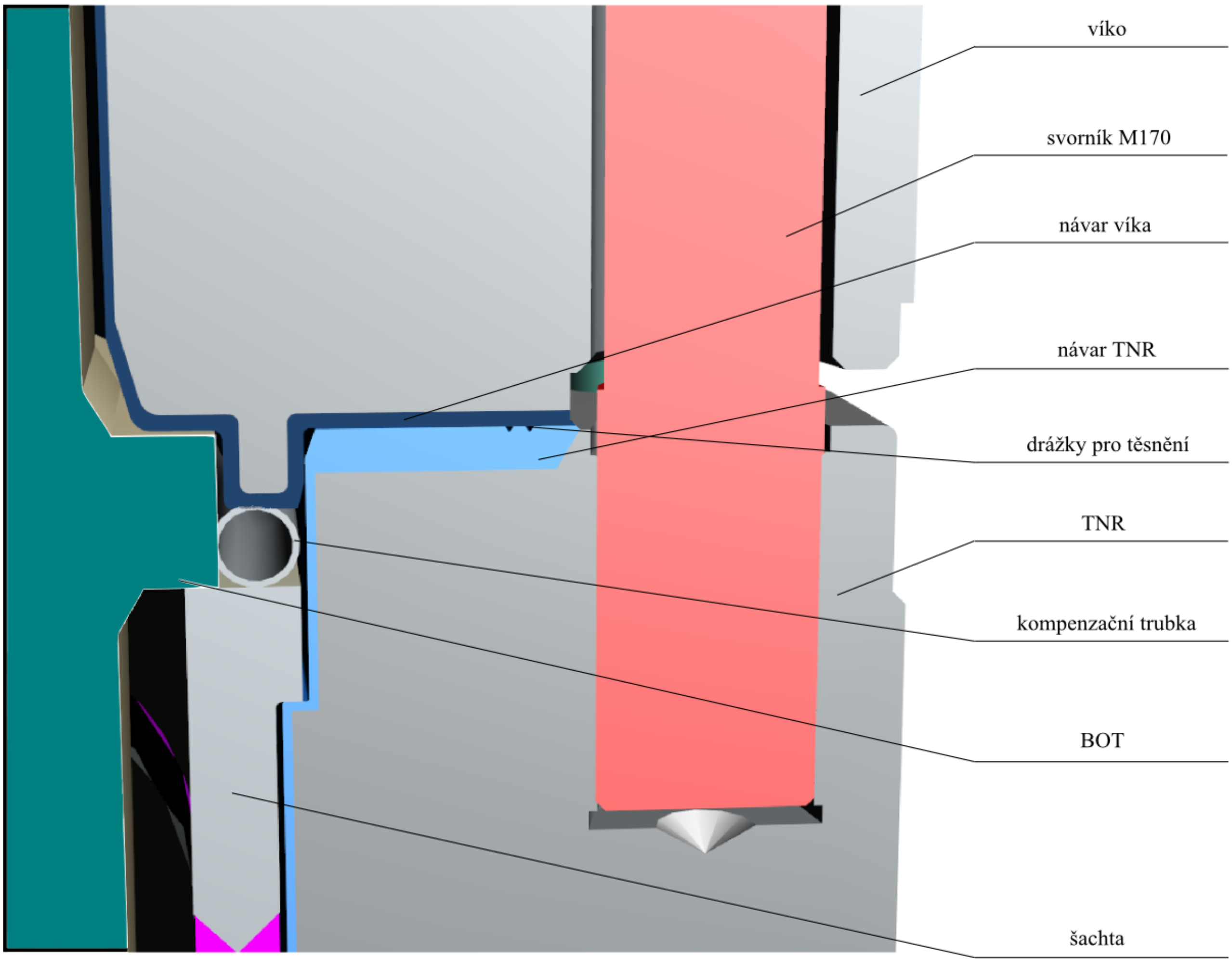


Fig. A.10: Flange - reactor parting line detail

Legend

Víko	Vessel head
Svorník M170	Stud M170
Návar víka	Head cladding
Návar TNR	RPV cladding
Drážky pro těsnění	Grooves for seals
TNR	RPV
Kompenzační trubka	Expansion compensating pipe
BOT	Protective tubes block
Šachta	Cavity

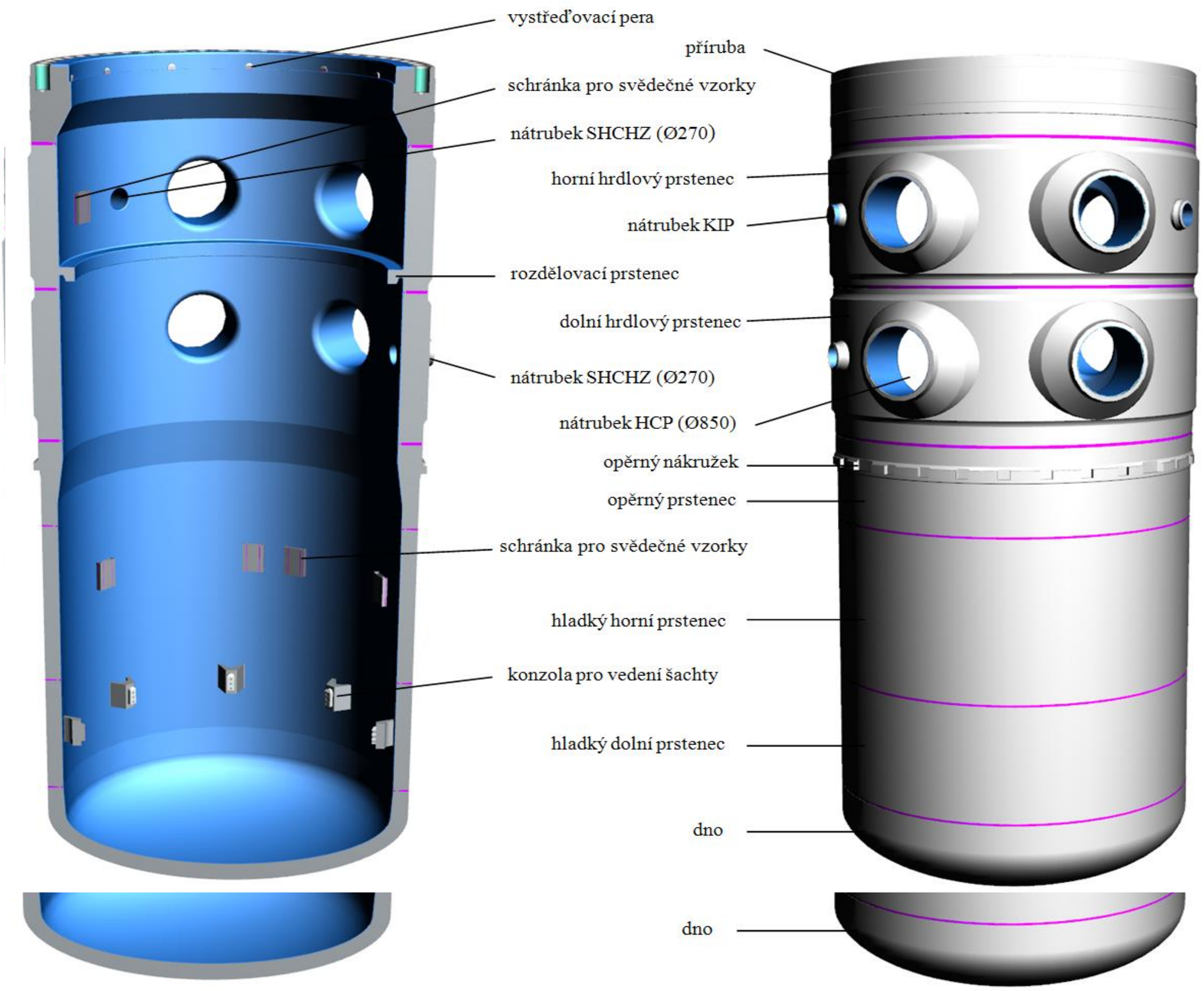


Fig. A.11: Reactor pressure vessel body

Legend

Vystřeďovací pera	Centering keys
Příruba	Flange
Schránka pro svědečné vzorky	Container for surveillance specimens
Nátrubek SHCHZ	ECCS nozzle
Horní hrdlový prstenec	Upper nozzle ring
Nátrubek KIP	I&C nozzle
Rozdělovací prstenec	Partition ring
Dolní hrdlový prstenec	Lower nozzle ring
Nátrubek HCP	Main circulation piping nozzle
Opěrný nákružek	Loose collar
Opěrný prstenec	Support ring
Schránka pro svědečné vzorky	Container for surveillance specimens
Hladký horní prstenec	Smooth upper ring
Konzola pro vedení šachty	Bracket for cavity guide
Hladký dolní prstenec	Smooth lower ring
Dno	Bottom

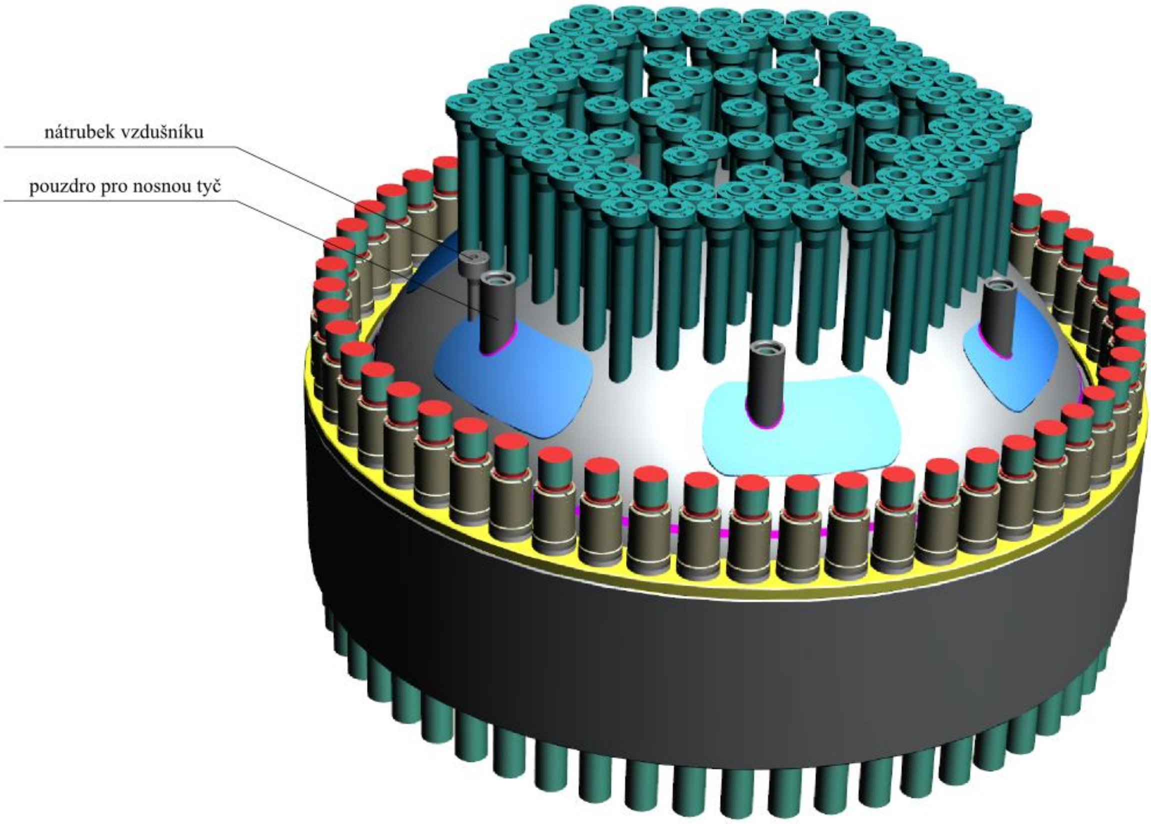


Fig. A.12: Reactor pressure vessel head

Legend

Nátrubek vzdušníku	Air tank nozzle
Pouzdro pro nosnou tyč	Bushing for support rod

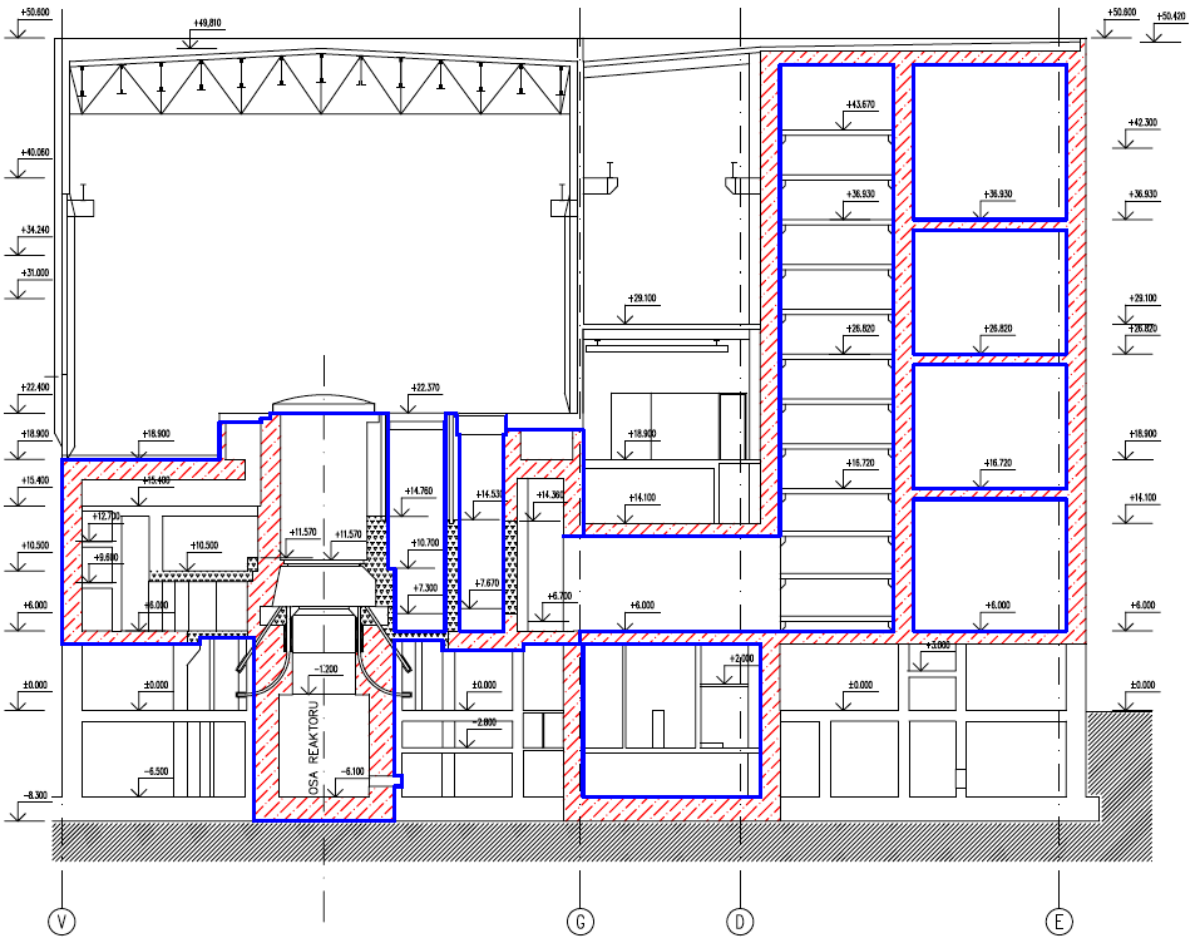
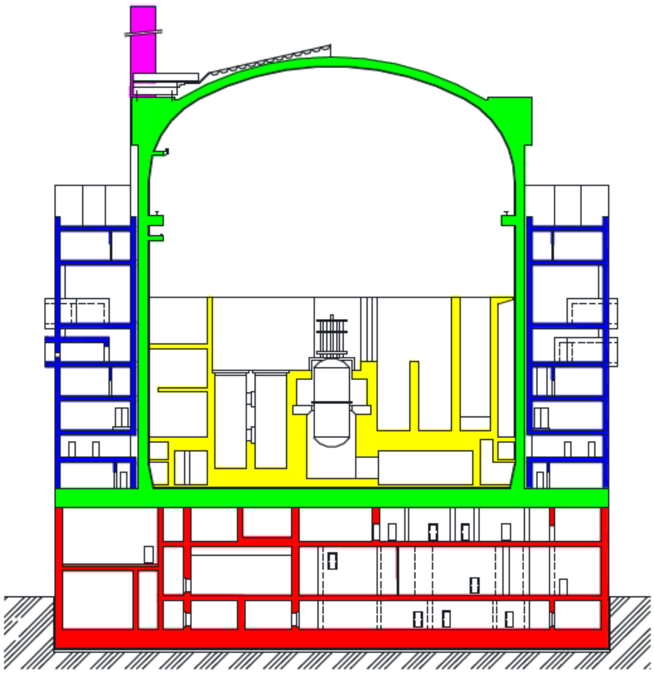


Fig. A.13: Layout of the reactor building of the Dukovany NPP



Dělení budovy reaktoru: základová část, hermetická část (kontejnment), hermetická část (vnitřní vestavba), obestavba, ventilační komín
část (vnitřní vestavba), obestavba, ventilační komín

Fig. A.14: Layout of the reactor building of the Temelín NPP

Legend

Reactor building structure: foundations, hermetic area (containment), hermetic part (internals), enclosure, ventilation stack

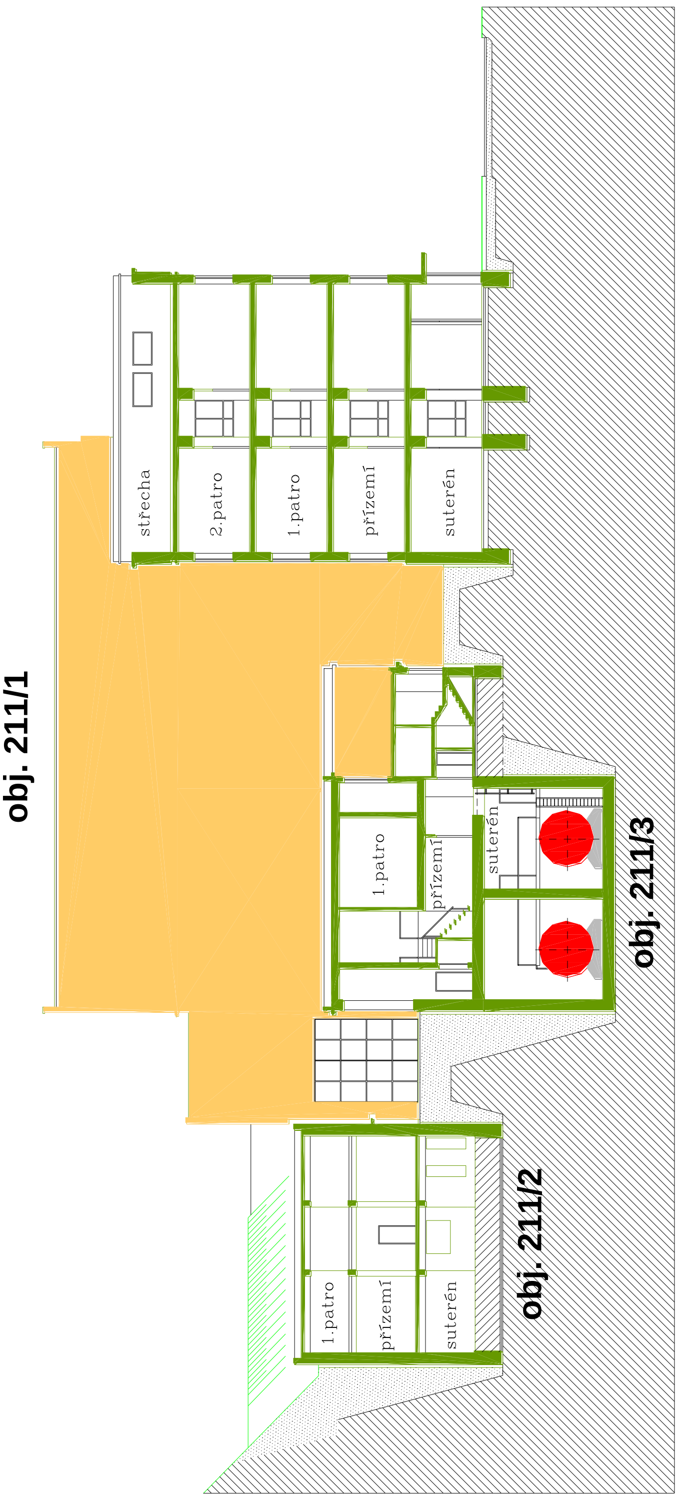


Fig. A.15: Layout of LVR-15 reactor hall

LVR-15 II.patro

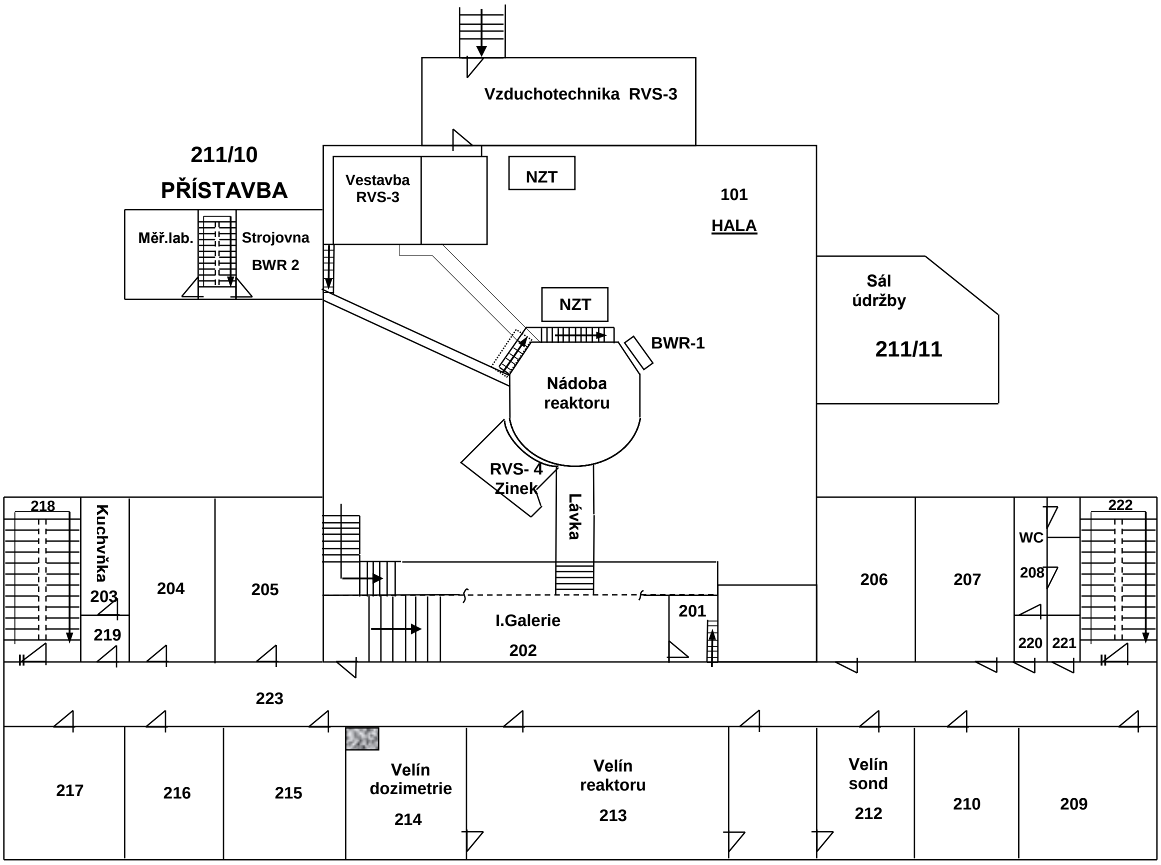


Fig. A.16: Layout of LVR-15 reactor hall

Legend

LVR-15 II.patro	LVR-15 floor 2
Přístavba	Annex building
Měř. Lab.	Measuring/testing laboratory
Strojovna	Machinery hall
Vestavba	Internals
Vzduchotechnika	HVAC
Hala	Hall
Nádoba reaktoru	Reactor vessel
Kuchyňka	Kitchen
Galerie	Gallery
Lávka	Footbridge
Sál údržby	Maintenance hall
Velín dozimetrie	Dosimetry control room
Velín reaktoru	Reactor control room
Velín sond	Probe control room

Annex B:

Table B.1: Summary of the safety cables of the Dukovany NPP included in the AMPC

The list was drawn up on the basis of the SSK statement on 2 November 2016. Only small portion of the cables (highlighted in yellow) of the total amount is included in the WENRA requirement (AAR list, Chapter 0.3.1.1 [1]).

Cable	In operation from	On the AAR list	Item in the AAR
Types of cables under harsh conditions with the requirement for resistance in DBA			
CXFE-R(V)/LOCA	2002		
CXKE-R(V)	2002		
CXKE-R(V)/HELB	2011		
CXKE-R(V)/LOCA	2002		
CHKE-R(V)	2000		
CHKE-R(V)/LOCA	2002		
JCXFE-R(V)/HELB	2011		
JCXFE-R(V)/LOCA	2011		
JE-H(ST)H	2002		
JYTY	1985		
KPOBOV/T3	1987		
KPOSG	1985		
KSC	2002		
KX-1-1-F-V/LOCA	2002		
LiHFKFHQE-R(V)	2002		
NU-THXHCHX/LOCA	2002		
SiHGLCSi/N2GMH2G	2000		
SISIF	2012		
TKC (Mirion, USA)	2015	yes	NIS
VCXJE-V (Kabelovna Kabex, Czech Republic)	2015	yes	NIS
Types of cables under harsh (as well as moderate) conditions. Resistance in DBA is not required			
ANKOY	1985		
AYKCY (Kablo Kladno, CSSR)	1985	yes	HV cable
AYKY	1985		
BYFY			
CPDEX PV	2013		
CXFE-R(V)	2002		
CYAY	1985		
CYKY	1985		
CYLY	1985		
CH(X)KE	1996		
CHKCE-R(V)	1995		
JCXFE-R(V)	2002		
J-LIHH(St)	2002		

Cable	In operation from	On the AAR list	Item in the AAR
JXFE-R(V)	2000		
K-ALUMEL-V/LOCA	2000		
KMPEVE	1985		
KMTVEV	1985		
KPETI	1985		
KPOBOV	1985		
KPOESV	1985		
KVVGE	1985		
LiHKFHQE-R(V)	2002		
LYS	1985		
MK	1985		
NCEY	1985		
NSKB	2010		
PVSG (SSSR)	1985	yes	HV cable
SHFKHFHQE-R(V)	2002		
SHKFHQE-R(V)	2000		
TCEKFY	1985		
Types of cables that occur only under moderate conditions			
(N)HXH-O	2015		
AMP	2010		
AYY	1985		
CBL300	2002		
CGAU	1985		
CGSG	1985		
CGTG	1985		
CMFM	1985		
CMSM	1985		
CNKOY	1985		
CXKCE-R(V)	2000		
CXKFE-R(V)	2000		
CXKH-R	2002		
CYA	1985		
FTP CAT.5E	2015		
HSLCH	2014		
CHAH-R(V)	1985		
CHBU	2014		
CHFE-R	1995		
CHKFE-R(V)	2014		
CHTH-R(V)	1995		
J/A-DQ(ZN)HH	2015		
JCXFOE-R(V)	2002		
JQTQ	1985		
JZ500	2014		

Cable	In operation from	On the AAR list	Item in the AAR
KEFS	1985		
KJB	2010		
Koax RK 75	1985		
KSB	2000		
KUGVEV	1985		
KUHSB	2010		
KX-1-1-F-R	2000		
LAN 1A	2004		
N05Z1Z1-K	2010		
N2XH	2010		
NCYY	1985		
PAARTRONIC	2002		
Pirelli CP(Prysmian, France)	2002	yes	NIS
RADOX	2000		
SCXFOE-R(V)	2000		
SHKFE-R	2000		
SY	1988		
SYKFY	1985		
TCEKE	1985		
YY	1985		

Table B.2: Summary of the safety cables of the Temelín NPP included in the AMPC

The list was drawn up on the basis of the SSK statement on 2 November 2016. Only small portion of the cables of the total amount is included in the WENRA requirement (AAR list, Chapter 0.3.1.1 [1]).

Cable	In operation from	On the AAR list	Item in the AAR
Types of cables under harsh conditions with the requirement for resistance in DBA			
KJA	2001		
KJTA	2001		
NSKA	2001		
NSKFA	2001		
NSKJA	2001		
WEC1031210	2001		
WEC1031211	2001		
Types of cables under harsh (as well as moderate) conditions. Resistance in DBA is not required			
C5XKE-R(V)	2007		
CXKE-R(V)	2001		
CXKE-R(V)/LOCA	2006		
EUPEN TXCR/2	2001		
CHAH-V	2001		
CHFE-R/LOCA	2006		
CHKE-R(V)	2001		
CHKE-R/LOCA	2006		
CHTH-R(V)	2001		
JCXFE-R(V)	2007		
JCXFE-R(V)/LOCA	2015		
KJB (Alcatel, France)	2001	yes	NIS
KJC	2001		
KSA	2001		
KSB	2001		
KSC	2001		
KSD	201		
KUHS	2001		
KUHSC (Alcatel, France)	2001	yes	HV cable
NSKB	2001		
NSKC	2001		
NSKJB	2001		
NSLB	2001		
NSLC	2001		
RADOX	2001		
Types of cables that occur only under moderate conditions			
2020206-WEC	2001		
2090399-Alpha Wire	2001		

Cable	In operation from	On the AAR list	Item in the AAR
2090999-VGA cable	2001		
4010304 WEC	2001		
9010220 - WEC	2001		
AMP	2013		
AMP data	2013		
AMP fibre-optic	2013		
C5HKE-R(V)	2001		
C5XFE-R(V)	2011		
CGLG	2001		
CGTU	2001		
CXFE-R(V)	2001		
CXKH-R(V)	2001		
H07V-K 10	2014		
H07V-K 16	2014		
CHBU	2015		
CHFE-R	2001		
CHKCE-R(V)	2001		
JXFE-R(V)	2001		
JZ-500 HMH	2014		
KJD	2001		
KJFB	2001		
KJSD	2001		
KJTB	2001		
KJFD	2001		
KOAX SRG8/U	2001		
KPETI	2001		
KUHSB	2001		
NFKB	2001		
NSFKD	2001		
NSKJD	2001		
NU-2XSEH	2014		
PRAFLASAFE X	2000		
S5XFE-R(V)	2008		
S5XKE-R(V)	2010		
SCXFOE-V	2010		
WEC 3A98892H02	2001	yes	NIS
WEC 406A066H01	2001		
WEC 406A066H02	2001		
WEC 406A100H01	2001		
WEC 406A100H02	2001		
WEC 4A06390H01	2001		
WEC 4A06390H02	2001		
WEC 4A07459H01	2001		
WEC 4A07467H01	2001		

Cable	In operation from	On the AAR list	Item in the AAR
WEC 4A07469H01	2001		
WEC 4A07470H01 (Chromatic Technologies, USA)	2001	yes	NIS

LAST PAGE

SÚJB

Státní úřad pro jadernou bezpečnost