National Report of Hungary

Topical Peer Review 2017

Hungarian Atomic Energy Authority

Hungarian Atomic Energy Authority
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01. General information

01.1 Nuclear installations identification

Key parameters of Hungarian nuclear facilities:

- Name: Paks Nuclear Power Plant (Paks NPP)
- Licensee: MVM Paks Nuclear Power Plant Ltd.
- Type: VVER-440 (V-213) – 4 units
- Rated power: 500 MWe / unit
- Start-up date: 1982, 1984, 1986, 1987
- Planned shutdown dates: 2032, 2034, 2036, 2037

- Name: Budapest Research Reactor (BRR)
- Licensee: Centre for Energy Research of the Hungarian Academy of Sciences
- Type: light-water cooled and moderated, tank-type reactor, with Beryllium reflector and with VVER-SM (-M2) fuel
- Power: 10 MWth
- Start-up date: 1959
- Planned shutdown date: not determined, current operation license is valid until 2023.

- Name: Training Reactor
- Licensee: Budapest University of Technology and Economics
- Type: light-water cooled and moderated, tank-type reactor, with graphite reflector and with EK-10 fuel
- Power: 100 kWth
- Start-up date: 1971
- Planned shutdown date: not determined, current operation license is valid until 2027.

According to the specification developed for the Topical Peer Review (TPR) process, two nuclear installations belong to the scope of the review: the Paks NPP and the Budapest Research Reactor. In the National Assessment Report these two facilities are outlined.

01.2 Process to develop the National Assessment Report

Based on the Nuclear Safety Directive [2014/87/EURATOM (NSD)] of the European Union the member states are obliged to conduct a topical peer review in nuclear facilities from 2017 every 6 years. The basic goal of the TPR in 2017 is to review the ageing management (AM) practice of the nuclear facilities of the participating member states in order to identify good practices and areas for further development. The topic of the review has been determined by ENSREG.

In Hungary it is the task of the Hungarian Atomic Energy Authority (HAEA) to coordinate the peer review process. As first step a related national self-assessment was prepared, its results are compiled in this National Assessment Report (NAR). The WENRA has prepared a specification [A0] for the review, in which it described the background and scope of the review in 2017 and specified the requirements on the content of the NARs.

Upon this the HAEA developed its specification on the national assessment [A1] in which it adapted the WENRA specification, i.e. it specified the facilities concerned and the chapters of the NAR to be developed, distributed the tasks among the concerned Licensees and the HAEA, determined the overall schedule of the review for both the Licensees and the HAEA.

In order to execute the tasks, as first step the HAEA informed the two concerned Licensees in an official letter [A2] that they should carry out the Topical Peer Review in 2017. On 24 January 2017 the HAEA organized a kick-off meeting with the participation of the representatives of the Licensees. In the kick-off meeting HAEA gave information about the background of the review and the schedule of the tasks related. Based on the discussions in the meeting the HAEA finalized the specification and officially sent it to the Licensees and requested them to carry out the review upon the specification.

The HAEA conducted formal inspections in June 2017 at both Licensees to confirm whether the topical peer review process was taking place according to the specification and the Licensees would complete their self-assessment by the designated deadline. In the case of the Budapest Research Reactor the inspection concluded the need of a more intensive work to be able to complete the respective part of the NAR by the Licensee. External partners were not involved in the review. Paks NPP informed the HAEA during the inspection that it had involved its TSO (VEIKI Energia+ Ltd.) and its permanent technical consultant (Trampus and Co Ltd.) in the process for a preliminary expert review of the results. The work at the NPP was carried out according to the schedule. The HAEA took records of both inspections [A3, A4].

In line with the HAEA specification, the Licensees performed the self-assessment independently of each other, the Paks NPP with the inclusion of TSOs, then they concluded their designated chapters for the NAR, which were sent to the HAEA for preliminary review and then in a finalized form already considered the HAEA comments. The NAR has been compiled by the HAEA from the chapters and information prepared by the Licensees and itself. The final version has gone through a security information check to avoid publication of sensitive information. Finally, the English translation was prepared. The HAEA published the NAR in both languages with a press release in its public website (www.haea.gov.hu) indicating that questions and comments would be welcome to the report.

02. Overall Ageing Management Programme requirements and implementation

02.1 National regulatory framework

At the time of construction and commissioning of the concerned nuclear facilities basically there were no requirements in the national legislation on the comprehensive, systematic ageing management. At the time of commissioning of the NPP, there existed mandatory requirements on the design and operation of certain plant programmes (maintenance, in-service inspections, surveillance activities) which can be nowadays regarded as parts of an effective ageing management programme.

Ageing management as a concept appeared in the national regulatory system in 1995, when a guideline attached to a resolution of the HAEA [A5] was published on the method of implementation of the first Periodic Safety Review of Units 1 and 2 of Paks NPP. It contained detailed description how the systematic activity aimed at managing the ageing of the systems, structures and components (SSCs) should be evaluated and reviewed. The guideline was based on the international recommendations and national considerations, experience of that time. Its scope covered all safety related SSCs. The demand for implementation of an optimal and coordinated programme and most of the attributes of the currently regarded effective ageing management appeared in it.

The requirements reached the level of legislation in 1997 [A6], when by the government decree on nuclear safety requirements evaluation of ageing and ageing management activities became mandatory elements of the periodic safety review of all nuclear facilities [included the two nuclear facilities concerned] as well as it specified aspects how to do that. The need for investigation of ageing issues also appeared regarding nuclear safety related events and the design and operation aspects of ageing management were also included in the decree, though not on a systematic basis as it is expected today.

In the beginning of the 2000s, after that the owner of Paks NPP declared the extension of the design lifetime as a strategic goal, the HAEA elaborated the respective nuclear safety requirements and published them at first in regulatory guidelines, then in 2005 in a new government decree containing the revised nuclear safety requirements [A7], which already meant a system of ageing management meeting the requirements of today.

At the time of the preparation of the NAR, the nuclear safety requirements in force are included in the Govt. Decree 118/2011 Korm. [A8] and in its annex, in the Nuclear Safety Code (NSC). The NSC of 10 Volumes and the related guidelines on the implementation contain the detailed requirements on the design of ageing management and the operation of a comprehensive ageing management programme as described below:

1. Definition of *ageing, ageing processes, ageing management* and *ageing management programme* is in line with the interpretation of the WENRA TPR specification [A0] (Definitions 131-133 of NSC Volume 10).
2. Ageing processes to be expected throughout the lifetime shall be taken into account during the design (for Paks NPP: NSC 3.3.2.0200, for BRR: NSC 5.2.2.4200.).
3. Ageing processes shall be identified in the design, the data for ageing management programme shall be ensured, harmony with the maintenance and other plant programmes shall be provided, fulfilment of safety functions shall be demonstrated taking into account uncertainties and the lifetime of the SSCs shall be determined together with the indicators and criteria related to the ageing mechanisms. Design requirements on ageing management shall also be determined (NSC 3.3.2.3900-4400, NSC 5.2.5.0100-0200., 5.2.7.0200-0300, 5.2.10.0100.).
4. A comprehensive ageing management programme shall be operated coordinated with the other facility programmes (maintenance, surveillance, qualification) including the establishment and maintenance of an ageing management database. The programme shall be reviewed and updated on a regular basis. (NSC 4.6.2.0100-0700, NSC 5.3.9.0100-0600). In the case of the nuclear power plant further stipulations are applied to the plant programmes: maintenance effectiveness monitoring shall be applied for active components, environmental qualification shall be applied for electric and I&C components in harsh environment, individual ageing management programmes shall be

developed for main circulation loop and spent fuel pool components. (NSC 4.6.0.0100-0200.)

5. Evaluation of ageing processes and effectiveness of ageing management is part of the ten-yearly due Periodic Safety Review (NSC 1.7.3.0500., NSC 4.6.2.0700., NSC 5.3.9.0600.).
6. Regarding the NPP it is a precondition of the licensing of service life extension to operate an effective ageing management program, which shall also contain the review of ageing management of passive and long lived components. (NSC 1.2.6.1400, 4.15.0.0500-0700.).
7. Regarding the NPP, the HAEA published detailed guidelines for the Licensee on considering the ageing during design and implementation of ageing management in operation ([A9], [A10]).
8. Preparation of the NPP for the planned extension of design service life and for the implementation of its licensing are also supported by regulatory guidelines which contain the ageing management related requirements ([A11], [A12]).
9. Obligations on regular reports by the nuclear facilities contain that the experience related to the operation of ageing management programme should be described in the annual report according to the specified aspects ([A13], [A14]).

In the Hungarian regulatory system the use of technical standards is not mandatory, but can be referred in regulatory resolutions or during other regulatory procedures. On the other hand the Act on Atomic Energy [A15] stipulates that it is mandatory to use technical standards in supporting license applications of nuclear facilities. The NSC (NSC 2.2.1.0300) requires that the management system of the facility shall determine the international regulations and standards adapted by the Licensee. Design of SSCs shall take place according to the adopted standards recognised in nuclear industry (NSC 3.2.1.2100., NSC 5.2.22.0300.). The NSC refers to the use of technical standards in several provisions. Based on all these, the Licensee shall determine the standards to be used during ageing management and demonstrate their adequacy for use with the stipulation that only recognised standards can be used.

In Hungary the Act on Atomic Energy stipulates that a graded approach shall prevail in the application and oversight of atomic energy. Accordingly the nuclear safety requirements are more stringent, more detailed for the nuclear facilities representing a higher risk level, while the regulatory procedures are differentiated according to nuclear safety significance as required by the decree on the Nuclear Safety Requirements [A8]. The ageing management requirements described above follow, and the regulatory procedures applied to oversee this area consider this basic concept. The HAEA has also taken into account the graded approach in the assessments of the Topical Peer Review.

02.2 International standards

Paks Nuclear Power Plant

The well known ageing management (AM) system, described in the IAEA NS-G-2.12 Safety Guide[1] [1] shown in Figure 02.2-1, served as the principal basis for the overall ageing management programme of the Paks NPP.

[1] Still in force at the time of the compilation of this report.

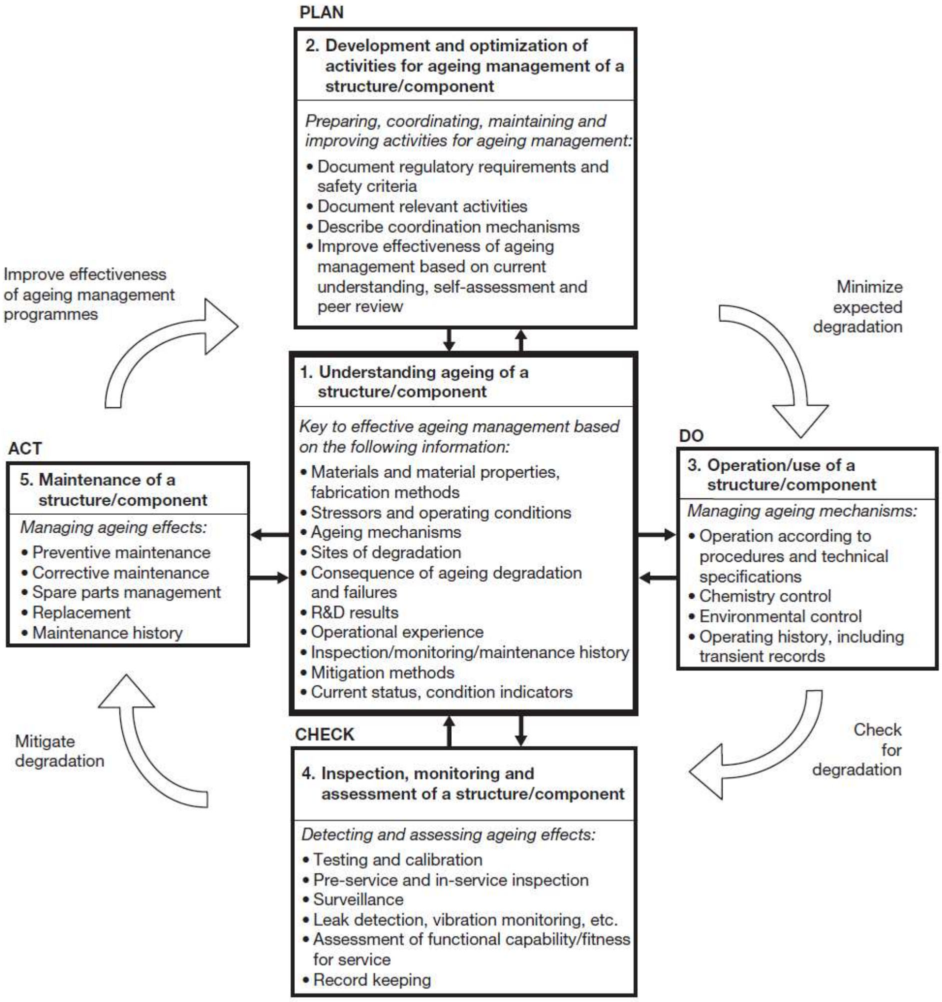

Figure 02.2-1: An example of a systematic approach to ageing management

Figure 02.2-1. is also included in the IAEA DRAFT SAFETY GUIDE DS485 [2], which is a revision of the IAEA NS-G-2.12 [1]. Although the new figure differs slightly from the previous version, the basic principles of the systematic approach to AM system remain the same.

The Overall Ageing Management Programme (OAMP) of the Paks NPP contains all the essential elements shown in the ageing management system figure above. The most important international standards used in their development, improvement and operation are the following.

- NUREG-1801 GALL [3],
- IAEA-TECDOC-1668 [4],
- IAEA-TECDOC-1557 [5],
- IAEA-TECDOC-1361 [6],
- IAEA-TECDOC-1025 [7],
- IAEA-TECDOC-1188 [8],
- IAEA IGALL [9] reports and
- WENRA [10] reference levels.

The Degradation Oriented Ageing Management Programmes (DAMP) focused on specific ageing mechanisms and their management as required by the national regulatory framework, as well as component Specific Ageing Management Programmes (SAMP) are based on the following 10 attributes:

1. Determination of degradation mechanisms, and ageing-critical locations;
2. Measures mitigating and preventing ageing mechanisms;
3. Parameters to be monitored;
4. Detection of ageing effects;
5. Monitoring, trending, condition assessment;
6. Acceptance criteria;
7. Corrective measures;
8. Feedback, improving efficiency of the ageing management programme;
9. Administrative control – Quality assurance, coordination, documentation;
10. Utilization of operating experience.

The ASME BPVC XI (2001) [11] standard (published as MSZ 27011 [12] Hungarian standard in 2013) has been used by Paks NPP:

- for the development of main programmes ensuring the detection of effects of ageing, i.e. the In-Service Inspection Programmes (ISIP) and of the material testing (NDT) framework programmes applied to those,
- for the definition of acceptance standards of the ISIP and NDT programmes, and
- for the development of repair technology of detected faults.

It ensured also the compliance of these programmes with international practices.

Budapest Research Reactor

At the time of launching of the BRR Ageing Management Programme in 2005, the *IAEA-TECDOC-792 Management of Research Reactor Ageing* publication was taken as a basis. During the subsequent review of the programme the *IAEA SSG-10 Ageing Management for Research Reactor* [A16] recommendations were used.

The physical and non-physical ageing effects and the steps of the ageing management process to be taken into account in the BRR were determined based on these reference documents.

02.3 Description of the overall ageing management programme

Paks NPP

02.3.1 Scope of the overall AMP

The scope of systems, structures and components (SSCs) important for nuclear safety was determined based on the national regulations – which is in line with the IAEA standards and international good practices. These SSCs are managed by ageing management, environmental qualification or maintenance and maintenance effectiveness monitoring (MEM) according to the following main groups:

- Ageing management is used for passive mechanical components, buildings and other structures and passive electrical and instrumentation and control (I&C) components.

- MEM is used for active components.
- Electrical and instrumentation and control (I&C) components, certain structural elements and most of cables are managed by using and maintaining an environmental qualification.

02.3.1.1. Responsibilities

The Company Organizational and Operational Rules [13] states that the overall ageing management is the responsibility of the Technical Support Directorate. The Directorate of Technical Support has delegated the authority to an organ with appropriate independence, the Ageing Management Section in relation to the following tasks:

- Management and supervision of technical activities carried out for ageing management;
- Development and publishing ageing management programmes;
- Supervision of the technical/scientific content of AM programmes;
- Carrying out or contracting research and development (R&D) activities related to understanding ageing mechanism.

The implementation of the AM programmes within the Directorate of Technical Support falls within the competence of the Department for Mechanical Engineering, Department for Electrical and I&C Engineering, as well as the Civil Engineering Sections.

The maintenance program for electrical and I&C components and the inspection programs for structures and buildings are also applied as part of ageing management.

Several operative AM programmes are used for the ageing management of mechanical components; among these, the most important are summarized in Figure 2.3.1.1-1. These are the following:

- In-service Inspection Programmes of pressure-retaining vessels, filters and heat exchangers among the priority and non-priority mechanical components:
  - The inspections specified in the material testing framework programmes and visual structural inspection programmes are performed in this framework along with the hydrostatic and other leakage tests.
  - The NDT programs cover the priority and the non-priority components.
  - Visual structural inspections – except for non-prioritized pumps and valves – are performed on all pressure-retaining mechanical components.
- Visual inspection is carried out also in the framework of the periodic maintenance programmes of priority and non-priority mechanical components, such as vessels, heat exchangers, filters, pumps, fans, and calorifiers.
- Visual inspection which is part of condition-based maintenance of valves.
- For components that are not in the scope of the operative programs, i.e. without periodic maintenance and checks, reference condition inspection is carried out.
- There are programmes for targeted inspection of a given degradation effect, such as:
  - the reactor pressure vessel surveillance programme that monitors the irradiation damage of the vessel material;
  - programme monitoring the load cycles of equipment and components critical for fatigue;
  - water chemistry programmes that serve for the prevention and mitigation of the corrosion of internal surfaces that are in contact with medium;
  - erosion-corrosion monitoring programme through which the wall thickness of pipelines is checked where this degradation mechanism plays a role.

- The leakage detection activities in flanged joints and walkdown inspection programmes used in the operational practice are also utilized to prevent or mitigate possible degradation effects.

Organizations responsible for the main operative ageing management programmes:

- In-Service Inspection Programmes – Directorate of Technical Support;
- NDT – Material Testing Section;
- Reactor Pressure Vessel Surveillance Programme – Material Testing Section;
- Civil engineering condition monitoring programmes – Civil Engineering Section;
- Maintenance programmes –Technical Section of the Directorate of Technical Support;
- Water management and water chemistry programmes – Chemistry Department;
- Maintaining required operating parameters, leakage monitoring, in-service walkdowns – Directorate of Plant Operation.

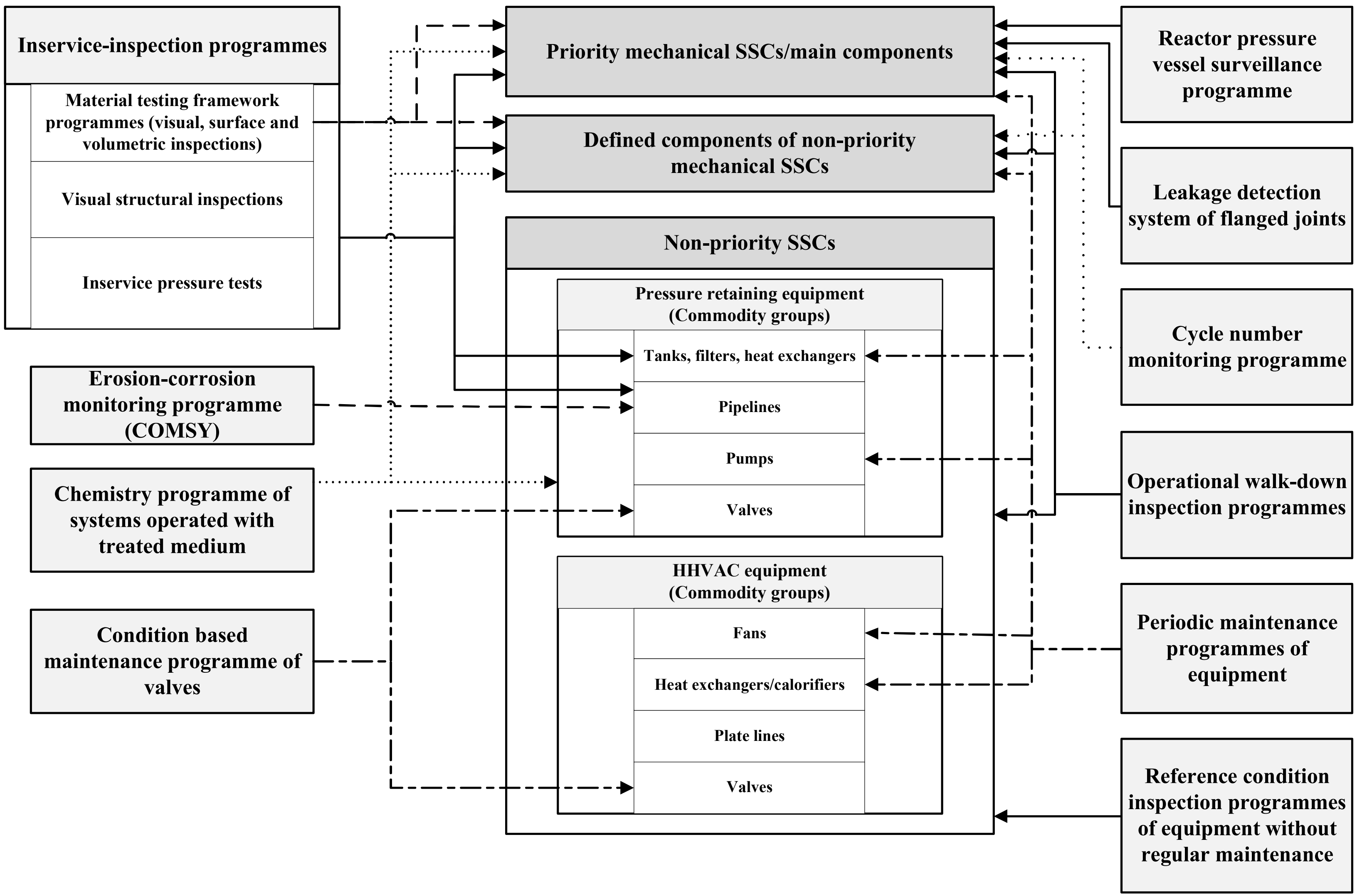

Figure 2.3.1.1-1: Main operative programmes associated with the ageing management of mechanical components in Paks NPP

02.3.1.2. Methods for identifying SSCs

As described in Section 02.3.1. ageing management is used in the case of passive components important to safety. Identification SSCs falling under the OAMP scope is carried out in accordance with the TBE304: Procedure of the Overall Ageing Management [14]. The parameters and considerations (nuclear and seismic safety classification, safety function, determination of active-passive components) required for the determination of the scope are contained in the central technical database (hereinafter AS6) of the nuclear power plant. The database of SSCs and cables covered by ageing management can be found in the AS6 as well as in the ADRIA information systems.

02.3.1.3. Grouping of SSCs in the screening process

Ageing management of the components within the scope of the AM is regulated by guideline 4.12 [15] issued by the HAEA. In the case of the components within the scope of AM, as well as for groups similar from AM aspect, the Guideline provides the development and operation of specific ageing management programmes (SAMP) containing the 10 attributes also required by the GALL report [3].

Mechanical components

Individual ageing management programmes have been developed to manage the ageing of high-priority major mechanical components.

In the case of non-priority mechanical components, the SSCs have been categorized into commodity groups that have the same ageing mechanisms and their management is similar. The development of these system component groups was defined by the identity of features such as the type of the system component, structural materials and the operating environment. In some cases, the establishment of additional subgroups within the SAMP was also needed, since certain degradation mechanisms or certain managing possibilities had to be/or could be taken into account on certain specific equipment or components.

There are also mechanical components that are treated by individual, component specific ageing management programmes.

The formulation of mechanical commodity groups as well as of the codes of the corresponding system component groups are defined based on the principles shown in Table 02.3.1.3-1. Annex 02.3.1.3-1 summarizes the ageing management commodity groups (SCGs) covering the ageing management scope of mechanical components, individually managed high-priority and non-priority components, as well as the corresponding SAMP IDs.

Table 02.3.1.3-1: Commodity groups and the formulation of the corresponding commodity group ID

Format of the commodity group ID: $AN_1N_1N_2$				
ID character	**A**	**N_1**	**N_1**	**N_2**
Meaning of characters	Type of equipment	Medium 1	Medium 2	Material of equipment
Possible character values	B – Tank D – Pump DV - Fan N – Filter S – Valve W – Heat exchanger Z – Piping, steel plate duct	1 – Primary circuit water 2 – Primary circuit steam 3 – Treated water 4 – Water steam 5 – Danube water 6 – Other impure solution 7 – Oil 8 – Gas 9 – Acid/Base 10 – Steam-gas mixture		1 – Carbon steel, non-corrosion-resistant steels 2 – Corrosion resistant steels 3 – Other alloys 4 – Different materials (Carbon steel, non-corrosion-resistant steels / corrosion resistant steels) 5 – Different materials (corrosion resistant steels / other alloys) 6 – Different materials (Carbon steel, non-corrosion-resistant steels / other alloys) 7 – Different materials (Carbon steel, corrosion resistant steels/ other alloys)

Buildings and other structures

For the ageing management of buildings and structures structural groups were formed in accordance with regulatory requirements and recommendations. Each structural group was categorized into main groups based on the type of material and the structural properties. The principle for organizing the groups was that elements with the same (similar) properties (material, construction, environment) should have the same (similar) degradation mechanism and ageing effects, therefore their ageing management would also be the same (similar).

For implementing the ageing management, structure-programmes and building-programmes are developed.

Most of the structure-programmes are related to one structural element main group or subgroup, while a smaller part of these (5 programmes) discusses the handling of certain specific phenomena and processes (e.g. machine base fatigue, reinforced concrete structures exposed to elevated temperature etc.), as it is shown in the table of Annex 02.3.1.3-2.

The building-programmes identify the structure-programmes relevant to specific complex buildings and determine the specific conditions, framework, logistical aspects, etc. of the implementation of the structure-programmes for each structural commodity group.

A relatively large number of structure-programmes (21 programmes) and building-programmes (32 programmes) are applied for the ageing management of buildings and structures.

Electrical and I&C SSCs

In the case of Paks NPP, the ageing management of most of the passive I&C components important to safety is realized through environmental qualification. To address the degradation mechanism of the passive electrical and I&C SSCs that do not have environmental qualifications, electrical and I&C system component groups that can be managed together from an ageing management aspect, have been arranged into the SAMPs listed in table 02.3.1.3-2 based on international and own operational experience.

Table 02.3.1.3-2: Ageing management programmes, commodity groups of electrical and I&C SSCs

SAMP ID	System component groups managed in the SAMP
V-SAMP-01	6 kV electrical distribution busbar, support insulators and electric connectors
V-SAMP-02	0.4 kV DC rigid busbars, conductor connection clamps and electric connectors
V-SAMP-03	Small distributor busbars, insulators and electric connectors (HIMEL CRN and RITTAL type enclosures)
V-SAMP-04	Electric connectors of PRISMA type sub-distributor equipment
V-SAMP-05	Cast aluminium and plastic busbars, insulators and electric connectors
V-SAMP-06	Wired-wound electric connectors
V-SAMP-07	Low-voltage PVC insulated cables
V-SAMP-12	Cables without environmental qualification
V-SAMP-13	Measurement cables without environmental qualification

02.3.1.4. Evaluation of maintenance practices to develop new ageing management programmes

While assessing the adequacy of the maintenance practice from AM point of view, the broad interpretation of maintenance is considered as the starting point in accordance with TBE304: Procedure of the Overall Ageing Management [14]. Before the finalization of new ageing management programmes, prior to the adoption of amendments to existing ones, the ageing management activities, documents governing the operative AM programmes must be amended, or new ones need to be developed. The relevant documents are:

- In-Service Inspection Programmes (ISIP);
- Material Testing Framework Programmes [16];
- Two Level Acceptance Standards (TLAS) [17];
- NDT technologies;
- Maintenance instructions;
- Operational programmes (operation instructions/technology instructions; chemistry programmes, monitoring of number of cycles etc.);
- Maintenance of technical condition and replacement programmes.

When assessing the compliance of operative programmes with AM requirements, firstly it is necessary to identify those existing programmes that can be linked to the 10-attribute management of significant degradation mechanisms in the SAMP. Identification must specify the compliance of the operative programme as well as additions and modifications required for it to be compliant. If there aren't any operative programmes complying with the AM needs of a given component, new programmes are developed for completing existing AM programmes. Such programmes are for example the condition assessment programmes of selected reference components.

The applicability of operative programmes with AM needs must be examined even if the components have experienced unexpected degradation mechanisms, or are experiencing the acceleration of suspected degradation effects. In such cases, changes can be made to the scope, frequency of the AM programmes, and/or more advanced test methods can be introduced or carried out.

02.3.1.5. Quality assurance, efficiency assessment

Quality assurance is ensured through the integrated management system in accordance with the Management Manual [18], internal regulations related to the implementation of ageing management programmes, procedural requirements and their fulfilment. Data collection, trend assessment of maintenance and operational history required by the ageing management programmes are carried out according to TBE304 [14] and the operation of AM programmes is carried out as specified in TBE305: Procedure for AMP operation [19].

- Inspections, reviews and maintenance activities are carried out in accordance with the scope and method provided in the AMP. Evaluation and documentation of the results in AS6 takes place based on the provided criteria.
- In the case of priority mechanical equipment, the ageing management engineer, in the case of other ageing management programmes the operative ageing manager checks the ageing management results and evaluates the operation of the AM programmes.
- If the ageing management criteria are not met or undesirable trends occur, upon completion of necessary corrective actions, the measures, modifications of the AMP, repairs, replacement must also be documented in the AM database.

Indicators related to plant procedures required by the provision of 2.5.1.0300. of the Nuclear Safety Code (NSC) are prepared for the TBE304 [14] and TBE305 [19]. Three indicators are used in the annual AM report for assessing the effectiveness of the relevant AM activities. These include:

- number of new degradation mechanisms and/or degraded locations found during maintenance and inspections;
- quality and quantity of feedback notes received from the HAEA after evaluation and acceptance of different AM documents;
- ratio of the AMPs implemented and planned.

The effectiveness of the OAMP should be evaluated annually based on the implemented ageing management activities. The scope and formal requirements of the evaluation are set out in Guideline 1.24 [20] issued by the HAEA for the annual AM report. The comprehensive assessment of the effectiveness of the OAMP, the comprehensive assessment of possible trends also takes place every 10 years during the Periodic Safety Review (PSR) in accordance with Guideline 1.39 [21] issued by the HAEA.

02.3.2 Ageing assessment

02.3.2.1. Consideration of documents

As described in Section 02.2, to develop the main elements of the OAMP, the most important international standards, guidelines and documents were used. Degradation oriented ageing management programmes (DAMP) and commodity group specific ageing management programmes (SAMP) must be developed in accordance with the requirements set out in the Guideline 4.12 [15] issued by the HAEA. Operative AM programmes should also be adjusted to the operation of SAMPs.

The available manufacturing documentation, such as drawings, manuals, technical specifications, production quality assurance data, welding specifications, strength, brittle fracture and fatigue calculations are used for the technical basis of the SAMPS and operative AM programmes.

The most important operational documents used in the development of the SAMPs and the operative AM programmes were the Operational Limits and Conditions of Units 1-4 (OLC) [25] and the system specific operating instructions.

02.3.2.2. Identifying ageing mechanisms and their consequences

For a significant part of the SSCs within the scope of the AM, the basis for identifying the degradation mechanisms to be managed is the degradation locations/degradation mechanisms listed in the annex of AM Guideline 4.12 [15]. To develop this AM guideline, the HAEA has developed background documentation for the ageing management of the components listed in the annex of guideline 4.12 [15] issued by the HAEA. At the request of the HAEA, an international expert group also reviewed the draft guidelines for AM, thus Guideline 4.12 [15], too; the HAEA considered the results of the reviews during the finalization of the guidelines.

Elaboration of the SAMPs, identification of degradation locations and degradation mechanisms, as well as ageing effects not listed in the annex of Guideline 4.12 [15] issued by the HAEA, was based on own and industry experiences, and on relevant international requirements and recommendations (such as NUREG 1801 [3], IAEA-TECDOC-1556 [26], IAEA-TECDOC-1668 [4], IAEA-TECDOC-1557 [5], IAEA-TECDOC-1361 [6], IAEA-TECDOC-1025 [7], IAEA-TECDOC-1188 [8], IGALL [9]).

The TBE304 procedure [14] states that the operational experience of ageing management programmes, based on industry practice may require:

- management of new degradation mechanisms;
- clarification of the technical-scientific knowledge about the degradation mechanisms;
- exclusion of the occurrence of a degradation mechanism (due to replacement and/or effective preventive measures).

In such cases, a new DAMP may be developed, or the existing DAMPs can be modified or cancelled.

02.3.2.3. Definition of acceptance criteria

Ageing management programmes record the permissible degree of ageing effects due to a degradation mechanism and/or the method of determining acceptance criteria based on the expected sustainability of the safety function of a given component.

In addition to the international practice and criteria (ASME BPVC XI (2001) [11], NUREG 1801 [3], IAEA IGALL [9], IEEE 323 [27] etc.) used to determine the acceptance criteria describing the effect of ageing and necessary to determine the technical condition of SSCs and the required/expected technical condition of SSCs, also the proven criteria from our own maintenance and inspection practices are applied. In some cases, the acceptance criteria are determined as a result of unique model calculations. Sections 03-07. of the Report present examples of direct and indirect acceptance criteria of ageing effects.

02.3.2.4. Key elements of the plant's programme in ageing assessment

Key elements of the ageing assessment are those operative AM programmes assigned to the SAMPs that ensure detection, monitoring of ageing effects (local/general material loss, occurrence and propagation of cracks, material property changes etc.) to ensure that the adequacy of the technical condition of the system component can be evaluated. The main programmes are:

- In-service material testing programmes (KA-01÷10_C15 material testing [16]), the acceptance criteria of which are contained in the Two-Level Acceptance Criteria [17], the criteria are the same as the ones contained in ASME BPVC XI (2001) IWB, C, D-3000;
- In-Service Inspection Programmes (including structural inspections with ASME BPVC XI VT-1 and pressure tests with ASME BPVC XI VT-2 visual inspections);
- RPV surveillance programme;
- Erosion-corrosion monitoring programme (COMSY – Condition Oriented ageing and plant life Monitoring SYstem);
- Visual monitoring programmes carried out under the maintenance programmes as well as during the condition assessments of structures, structural elements;
- Water chemistry programmes.

Sections 03-07. of the Report present examples of programmes associated with the detection of ageing effects, typical ways of using key power plant programmes.

02.3.2.5. R&D activities

A vital element of the OAMP is the definition, implementation and feedback of the results of R&D activities into the AM programmes. The determination and implementation of R&D activities required for AM is carried out by the Ageing Management Section by involving specialists responsible for a particular area. For the determination and external professional supervision of the more important or priority R&D activities, the Ageing Management Section also utilizes the Structural Integrity Scientific Advisory Panel (SI SAP) of the MVM Paks NPP Ltd.

MVM Paks NPP Ltd. is member of the EPRI and a participant in the NUGENIA programme. VVER related experiences of the projects and all-important elements of international practice of AM are implemented and utilized in the review of the AMPs.

Most important R&D activities related to ageing assessment:

- Stress corrosion sensitivity tests of the steam generator heat exchanger tubes;
- Irradiation examination of reactor pressure vessel wall;
- Assessments, tests, experiments related to the failure of the sleeves of penetrations of the Control Rod Drives Mechanisms (CRDM);
- Analyses, evaluations, experiments related to the damages of the main flange and thermal barrier of the Main Circulating Pumps (MCPs),
- Ageing management studies of cables;
- Condition assessment of reinforced concrete structures through sampling;
- Microbiologically influenced corrosion examinations.

02.3.2.6. Application of operational experience

An important source for identification of expected degradation mechanisms and ageing effects is the collection and evaluation of internal and external operating experience. Information is evaluated regularly, or if necessary individually, by the Ageing Management Section together with the sections of organization responsible for the operative AM programmes. AMPs are modified, amended as necessary, based on any new information. Evaluation of the experience gained in the scope of the SAMPs is carried out annually. The revision of industry information is also carried out annually. Modifications and additions required by the experience gained are added to the programmes during the periodic review of the SAMPs, however in the case of unique, substantive differences, immediate development and introduction of AM modifications may also take place immediately.

Examples of the utilization of internal operational experience and immediate modification of AMP are the modifications of the SAMPs and the operative AM programme because of the occurrence and management of corrosion damage in the cooling pipeline of the spent fuel pool. A typical example of modification in the AMP based on external operating experience is the identification and implementation of measures related to the management of the main circulating pump pressure head and guide wheel damage. Sections 03-07. of the Report present further examples on applying internal and external operational experiences.

02.3.3 Monitoring, testing, sampling and inspection activities

02.3.3.1. Monitoring condition indicators and parameters, trending

Monitoring and trending of the condition indicators of components important to safety (cracking, residual wall thickness, faultless condition, material property change etc.) as well as of the direct and indirect parameters related to ageing and ageing management (load cycles, changes of operational parameters, chemical parameters etc.) has already been going on since the commissioning of the plant. Coordination of these activities, the need for their utilization in a unified system was recognized in the first PSR and their implementation was realized in the framework of the introduction and operation of the OAMP, at that time. During the design and implementation of the OAMP, evaluation of these activities and their alignment to the ageing management needs were carried out. Sections 03-07. of the Report provide specific examples of monitoring and trending of condition indicators and parameters.

02.3.3.2. Testing programmes

Examinations required in the material testing framework programme, in the visual inspection structural testing programme and in the periodic pressure tests are carried out in the framework of the In-Service Inspection Programme of the pressure retaining equipment and pipelines. The material testing framework programmes cover priority and some non-priority pipings, pipe components. Structural testing is done on practically all pressure retaining components, except for non-priority pumps and valves. Visual inspections are also conducted as part of the periodic maintenance programme of mechanical components and of the condition-based maintenance of valves. In the case of components outside the scope of the above mentioned operative programmes, i.e. components without periodic maintenance or inspection, reference components have been selected and their condition testing is performed.

Condition assessments indicated in the AM programmes of electrical and I&C components are carried out within the maintenance programmes.

Examinations required in the AM programmes of buildings and other structures are carried out in the periodic condition inspection programmes.

Among the above-mentioned programmes, the activities of third party certified organizations may typically be linked to certified material testing of the material testing programme. Qualification is carried out based on the ENIQ (European Network of Inspection and Qualification) methodology directives and recommended practical guidelines as well as the IAEA's qualification methodology developed for VVER type nuclear power plants.

02.3.3.3. Surveillance programmes

The surveillance programmes of the plant are as follows:

- RPV surveillance (details in Section 05);
- Integrated leakage test programme of hermetic compartments (details in Section 07);
- Periodic sampling of the reinforced concrete structure of the containment surveillance programme (details in Section 07);
- Settlement examination programme of buildings (details in Section 07);
- Automated Ball Indentation test (ABI) on specific components of major equipment, through which the possible changes in the mechanical properties of the components concerned are followed.

02.3.3.4. Requirements for unexpected degradation mechanisms

Management of unexpected degradation mechanisms as set out in the OAMP core document [28] is as follows:

- in case of deviations revealed during condition review, event investigation, the possibility of new ageing mechanisms must also be considered;
- if necessary, the condition of similar components operating in similar environment must be checked;
- in the case of an unexpected degradation mechanism, or a degradation mechanism that is developing more rapidly than assumed, the process must be re-evaluated starting from the

design considerations and the necessary measures (e.g. modification of the SAMP) must be determined and applied;
- if the root cause is not clearly identifiable then at least all the consequences and the expected progress of the process must be established;
- a review should be carried out even if a deviation is not managed in the domestic practice, but occurs in international operational experience.

The requirements of the above mentioned sub-task and processes are contained in TBE304 [14], TBE305 [19] as well as the documents governing the implementation of the operative AM programmes.

02.3.4 Preventive and remedial[2] actions

02.3.4.1. Preventive measures laid down in the ageing management programmes

The specific ageing management programmes indicate the measures, activities to prevent the effect of degradation mechanisms in a SAMP as well as the operative ageing management programmes related to prevention. The main typical preventive measures are the following:

Mechanical SSCs

- Irradiation damage reduction: use of such core configuration that results in reduced fast neutron fluence to the reactor pressure vessel wall (details in Section 05);
- Prevention of boric acid corrosion: prevention of major leakages through the leakage control programme (details in Section 05);
- Prevention of all corrosion processes and environmental fatigue: chemical programmes of systems operating with treated, controlled medium (water, oil);
- Prevention of corrosion: using protective coating and preserving their consistency, periodic cleaning of surfaces;
- Prevention of local corrosion: using approved auxiliaries for maintenance;
- Prevention of fatigue: keeping the number and properties of the load cycles causing fatigue within the limits and, if necessary reducing the cycles, loads through modifying the operation; reducing thermal stress by installing thermal sleeves;
- Prevention of loosening at bolted joints: using fixing welds during manufacturing/assembly;
- Prevention of maintenance wear: using special maintenance methods.

Buildings and structures

- Prevention of any damage to accessible surfaces: by operating the building AMPs and structure AMPs together, repairing minor damages identified during walkdowns;
- Prevention of damage to foundation: examining the chemical composition of the ground water for assessing aggressiveness;
- Prevention of damage to pipe and cable support structures, doors and hatches: protecting the painting and coating of the structures from damages resulting from maintenance, repair activities (e.g. welding) carried out in the surroundings;
- Prevention of damage to reinforced concrete structures and masonry: elimination or minimization of water and boric acid solution leaks, repair of concrete surface faults,

[2] In Hungary in the English terminology the word „corrective” is used instead of „remedial”. In the report therefore ’remedial’ is only kept in section titles required by the WENRA specification and rather ’corrective’ is used in the text.

repair of the damaged parts (based on expert's opinion). The operational experiences and the results of experts' investigations show that mainly the surface layer of the heavy concrete structures is damaged by the boric acid solution.

- Prevention of damage to steel structures: using measures to avoid environmental effects causing degradation: elimination of leakage of water and boric acid solution, repairing the coatings and fasteners;
- Prevention of damage to the hermetic lining: elimination or minimization of water and boric acid solution leaks, avoidance of mechanical impacts, performing the integrated leakage tests at reduced pressure levels;
- Prevention of damage to concrete structures exposed to elevated temperature: by proper maintenance of insulation of equipment operating at high temperatures; by inspections of cooling of high temperature penetrations, and proper maintenance of the cooling system;
- Prevention of damage to surface coatings: by maintaining temperature within the limits, and by maintaining the level of radiation at a sufficiently low level; by protection of the surface coating from aggressive chemical effects and unplanned mechanical effects;
- Prevention of damage to machine bases: by reducing the source of vibration if necessary through repair, and by maintenance or operational modifications; by eliminating any oil leaks through proper maintenance of the mechanical equipment; by installing vibration insulation between the machine and the machine base;
- Prevention of soil damage: by erosion protection through proper grassing, surfacing and their maintenance.

Electrical and I&C SSCs

- Prevention of damage to cables, cable connections: preventing mechanical damage through careful maintenance;
- Activities to prevent damage to busbars, distributors, insulators, and electric connectors:
  - Maintaining the required environmental parameters;
  - Removal of surface pollutants;
  - Preventing the penetration of vapours and pollutants;
  - Preventing overload or short circuits;
  - Reducing maintenance loads.

02.3.4.2. Remedial measures established in the ageing management programmes

Specific ageing management programmes indicate the measures and activities to address the effect of the degradation mechanisms within a given SAMP, and the operative AM programmes related to the improvement measures. The main corrective measures are the following:

Mechanical components

- Local repair of cracks and local loss of material using approved repair technology;
- In case the fatigue analysis results in CUF>1 for a specific component, additional monitoring should be introduced;
- Recovery of design status through repairs or component replacement;
- In case of appearance of cracks exceeding the acceptance standards, crack grow analysis;
- Elimination of boric acid leaks;
- Construction modifications: e.g. modification of vulnerable fasteners, substitutions with less sensitive material for the particular degradation mechanism;

- Plugging of heat exchanger pipes;
- Periodic cleaning;
- Modifying the frequency, methodology of the periodic condition reviews as required.

Buildings and structures

- In case of deviations from the acceptance criteria, carrying out calculations, sensitivity tests;
- In case of a more serious degradation mechanism, modification of the frequency of the inspections, additional – even sampling based – examination;
- Correcting errors uncovered during condition reviews;
- The necessary corrective measures must be determined and ordered based on the maintenance practices of other power plant and based on individual expert opinion.

Electrical and I&C SSCs

- Replacement of a cable line that was proven to be unsuitable;
- Removal of heat and corrosion stains;
- Repair of faulty joints using proper material;
- Replacement of damaged insulators, insulation;
- Replacement of damaged parts with originals;
- Repair and replacement of elements that are aged or are not in good condition;
- Replacement of aged wire connectors;
- Replacement of cabinet seals;
- Cleaning of surfaces;
- In the case of oxidative ageing of certain components (joints, wires), investigation of the cause of overload or short circuit.

Budapest Research Reactor

02.3.1 Scope of the overall AMP - Budapest Research Reactor

The ageing management programme of the BRR was launched in 2005 after the completion of the first Periodic Safety Review. The objective of the programme is to identify and to monitor all degradation mechanisms and altogether to safely operate the reactor until the end of its design lifetime. The activity consists of the following, well-separable processes:

- Data collection, monitoring, surveillance of building conditions;
- Evaluation. Monitoring of exceedance of limit values, preparation of wear and usage trends;
- Intervention. Preparation and execution of repair and replacement schedules;
- Documentation according to the quality assurance practice of the BRR;
- Reporting on the activity of the given year;
- Adapting R&D experience, revision of the programme.

The following physical and non-physical ageing mechanisms are in the focus of the ageing management activity or programme:

- Neutron and gamma radiation damage;
- Thermal ageing;
- Creep (geometry change);

- Stress (pressure) caused breakage, collapse;
- Motion caused displacement;
- High cycle fatigue;
- Low cycle fatigue;
- Wear;
- Erosion;
- Corrosion, galvanic cells;
- Biological degradation (loss of efficiency of ion-exchanger resins during shutdown),
- Development of technology, moral obsolescence;
- Deficiency of documentation;
- Human factor.

The scope of the SSCs considered in the ageing management programme was determined in line with the principles of the IAEA publications, as follows:

- Safety function, typically safety class 1-3;
- Maintenance is possible, not possible;
- Replaceable, non-replaceable;
- In-service inspection is possible, not possible during shutdown.

SSCs within the AMP:

- Large reactor vessel containing the active core and its valves;
- Control and safety rods;
- Nuclear measurement chains;
- Primary circuit pipelines with the main gate valves;
- Heat exchangers;
- Gravity tanks (passive safety cooling);
- Make-up water tanks;
- Iodine and aerosol filter cartridges;
- Filters of the accident recirculation ventilation;
- Batteries;
- Spent fuel storage tanks;
- Liquid radioactive waste tank;
- Technological compartments of the reactor building;
- Venting stack.

The buildings and structures enveloping the SSCs within the AMP are also parts of the programme.

Responsibilities

At the research reactor, the maintenance activity, the monitoring/management of ageing mechanisms and environmental qualification of the equipment that fulfil safety function in accident conditions are conducted separately, but an overlap can be identified in terms of certain examinations, inspections and testing activity, which are performed simultaneously or successively.

The head and members of the Ageing Management Organization are assigned by the reactor manager. The activities of the organization are performed as a part time job, in parallel to their main assignments. The organization is directly subordinated to the reactor manager. The activity is described in annual reports, which is the attachment of the group manager meeting to be held at the end of each year. The activity of the organization is performed in accordance with the Quality Assurance Rules of the BRR [A17]. The composition of the organization:

- Development group: 1 mechanical engineer, who is also the leader of the organization;
- Mechanical group: 1 person;
- Electric group: 1 person;
- Instrumentation and control group: 1 person;
- Radiation protection and irradiation group: 1 person;
- Software expert of the reactor measurement data system: 1 person;
- Person responsible for the preparation of campaign reports: 1 person.

Identification of SSCs

Identification of the SSCs in the programme takes place based on the alphanumeric identification system of the research reactor. Beyond that the basic documents of the AMP display the safety classification and the safety function of the given component according to BIOS.

Grouping, screening of the SSCs

Grouping of SSCs in the programme and the application of screens is not necessary due to the simple construction of the research reactor.

Evaluation of the maintenance practice during the development of the new ageing management programmes

During the establishment, review of the comprehensive ageing management programme the results of the maintenance practice are fed back and considered.

Quality assurance, evaluation of efficiency

The measurement records are prepared on a unified template, the tasks are assigned on a Work Instruction Form standardized at the research reactor. Evaluation and archiving take place also based on the Quality Assurance Rule. The research reactor switched over to electronic use and archiving of data bases and measurement records step-by-step from 2012. Paper based documents are only used when indispensable.

Contractors are used where necessary to perform the tasks professionally. Qualification of contractors takes place according to the Quality Assurance Rule Sz-1.14. v.2. [A17]. It is important from the aspect of quality assurance that alphanumeric marking of the SSCs are assigned to the safety classification according to BIOS and the safety functions.

02.3.2 Ageing assessment – Budapest Research Reactor

The following four methods are applied to avoid or mitigate the ageing processes at the research reactor:

- Prevention, by which the ageing processes can be avoided. Such can be e.g. passivating, coatings, painting.
- Mitigation of ageing processes. E.g. increasing the frequency of water sample analysis, application of cascade filtering technologies (reverse osmosis procedure, HEPA filters).
- Adapting new diagnostic technology. E.g. measurement of wall thickness, ultrasonic examinations, under water videos.
- Performance monitoring. E.g. performance of heat transfer of heat exchangers, monitoring of capacity change of batteries, regular inspection of filtering efficiency of air filters.

As a first step, the selection of the SSCs and determination of the ageing processes take place.

The second pillar of the programme is the monitoring, data collection activity. This is described in detail in Section 02.3.3.

Evaluation of the data bases is the third pillar. In the case of commercial products the basis of the activity is provided from the manufacturers' requirements. In the case of individually designed equipment the technical design documentation, more specifically the detailed design and the transfer documentation is the basis. Regulatory guideline for ageing management only exists for nuclear power plants, which is not applicable to research reactors.

Depending on the data obtained:

- As a first step, monitoring of the ageing mechanisms detected at the SSC, determination of the extent of degradation and the demonstration of compliance of the SSC take place.
- Analysis of causes of the failures.
- Scheduling of repairs and replacements takes place by a coordination with the reactor schedule and the annual maintenance plan.
- Post-replacement examination and evaluation of replaced SSC condition.
- Feedback: utilization of experience regarding the activities, procedures and their modification, if necessary.

R&D activities

From 2009 the head of the AMP takes part in the IAEA Workshops and Technical Meetings. Two programmes are worth mentioning: IAEA Initiative on RR Ageing and Technical Meeting on RR Ageing Management organised by the IAEA and the Massachusetts Institute of Technology. The BRR does not conduct a systematic R&D activity meant to support its own ageing management, however on an occasional basis, if necessary, it performs such a research or contracts other organizations to conduct a specific research activity.

Using operation experience

Some of the original VVR-S reactors underwent significant modifications (power uprate, fuel conversion, etc.) and were shut down except for 2. There is one reactor in Vietnam at a power of 2 MWth and one similar in Ukraine. Technical condition of both of them is obsolete, the Ukrainian practically does not operate. It means that international experience with this type of reactors is not available.

Developments, improvement actions based on national experience:

- Visual and eddy current testing of control rods;

- Modification of spent fuel storage from wet to semi-dry storage;
- Modernization of venting system;
- Extension of water sampling;
- Modification of material of sealing of the primary circuit main gate valves;
- Modification of mechanical filter of the secondary circuit to ion exchanger type;
- Replacement of battery chargers to smart chargers with intelligent electronics;
- Modernization of the make-up water production equipment.

02.3.3 Monitoring, testing, sampling and inspection activities – Budapest Research Reactor

Monitoring of condition indicators and parameters, trending

270 parameters are recorded in Excel table from the measurement data collection computer, operational logs and forms during every campaign.

Main monitored parameters:

- Rotating machine operational hours;
- Operational parameters per reactor campaigns and summary data;
- Primary and secondary flow rate and pressure;
- Air flow rates and depression data;
- Tank water levels;
- Vibration measurement data of rotating machines;
- Water sample analysis results in terms of corrosion and fission product grouping;
- Falling times of safety rods.

Monitoring frequency of parameters:

- Continuously collected parameters of the reactor measurement collection system;
- Weekly inspections (tunnel, liquid radioactive waste tank);
- Pre-startup reactor inspections;
- Monthly inspections (water sampling, battery cell voltage, leakage wells);
- Quarterly inspections;
- Measurements during annual main outage;
- Vessel structural and pressure tests based on Hungarian Standard or Technical Review Plan (In-service Inspection Programmes) → 4, 6, 8 years.

Concerning trending the data from the Excel table can be used as appropriate from any parameters for any interval. Annual trending of battery cell capacity and the trend of the heat transfer of heat exchangers and cooling towers taken first for the PSR are worth mentioning.

Inspection and surveillance programmes

- Vessel structural and pressure tests according to Hungarian Standard and Technical Review Plan (In-service Inspection Programmes);
- Gamma dose rate measurement in wells to detect leakage from external spent fuel storage;
- Monthly pressure drop measurement across iodine and aerosol filters and gamma dose rate measurement;
- 10-yearly visual inspection of the reactor vessel;
- Annual capacity measurement of batteries;

– Annual manager circle of the buildings to review condition.

Management of unanticipated degradation processes

Management of detected degradation mechanisms always takes place based on individual management decision. Management of unanticipated events during the 12 years of AMP operation at the Budapest Research Reactor:

	Event	Management of event
1	Hole on the control rod	Replacement
2	Corrosion of battery electrodes	Replacement
3	Loosening of diffuser fixing bolts	Replacement
4	Embrittlement of primary circuit main gate valve sealing	Replacement
5	Corrosion of concealed portion of the secondary circuit pipelines	Replacement
6	Uncertainty of nuclear measurement chain	Replacement of chamber

02.4 Review and update of the overall AMP

Paks NPP

02.4.1. Management of audit and inspection results

The HAEA issued its most important observations and findings concerning the OAMP in relation to the service life extension programme belonging to the service life extension licensing of Units 1-4, the service life extension implementation programme and of the service life extension of Units 1-3 licenses. The HAEA also made comments in relation to the annual AM report.

Most of the HAEA's recommendations were connected to the scope of AM as well as to the missing technical basic data of components. Following the HAEA's review the scope of AM was adjusted, some programmes were amended. Technical basic data of components necessary for specific AM programmes were gained through supplementary data collection.

The HAEA reviewed the elements of ageing management of the important components identified in the construction review project. The HAEA's recommendations on the scope of these components were resolved by including the priority components in a separate procedure, too. The results of the required evaluation of the priority components are reported in detail in the annual AM reports.

02.4.2. Evaluation of operational experience

Evaluation of operation experience from AM aspects is ensured by the OAMP core document [28]:

- An obvious change in the trend of any of the parameters must be reported to the AM Section; the required analyses must be performed with the contribution of the AM Section.
- The organization responsible for investigating the event must notify the AM Section at the beginning of the investigation, who will consider the possible ageing degradation mechanisms concerned.

The crucial elements for evaluating operational experiences from AM aspects laid down in the OLC [25] are the loads, feedback on the decrease of load cycle numbers and summary report on chemical parameters to the Ageing Management Section in relation to the elaboration of the annual AM report.

02.4.3. Evaluation of the effects of modifications

Guideline 1.5 [29] issued by the HAEA on the modifications states that the effectiveness of programmes (AM, MEM, EQ, ISI) for ensuring continuous maintenance of the function of the modified or new SSCs has to be reviewed for the approval of the modification. Adequacy of the programmes after the condition change and the availability of the necessary baseline data shall be evaluated for licensing of the modification.

An obligatory part of the technical documentation for the modification is describing the AM aspects, where AM programmes related to modifications must be quoted. The AM programmes of the concerned SSCs must be included into the documentation presented for obtaining the approval.

For major modifications, such as a power uprate or the licensing of the 15-month operating cycle, a review, amendment and modification of the specific, degradation oriented and operative ageing management programmes for all SSCs in the scope of the modification had to be carried out, taking into consideration the changing operational, maintenance or other AM related conditions.

02.4.4. Review of AM, measuring and evaluating its effectiveness

02.4.4.1. Strategy of the periodic reviews

To maintain the required effectiveness of the OAMP it is essential to perform its periodical review the system, modify and supplement its elements as necessary, considering the current knowledge, experiences, regulations on ageing management.

The provisions concerning the periodic review of the specific and type ageing management programmes are contained in TBE304 [14]. The review of the AM programmes must be carried out at least once every 5 years. Based on the assessment of international and national experience, the review may take place more frequently. TBE304 [14] also displays the necessary modifications of the operative AM programmes' regulation documents related to the review of the specific and type programmes.

The comprehensive review of the ageing management – considering the content elements and requirements of Guideline 1.39 [21] issued by the HAEA – is also carried out during the PSR taking place every 10 years. The main elements of the review are: identification of the changes in national/international regulations related to AM, evaluation of the scope of AM, evaluation of the activities for the detection of ageing mechanisms and their developments, as well as the evaluation of the validity of the time limited ageing analyses (TLAAs) for the next 10 years.

02.4.4.2. Consideration of monitoring, inspection, testing and sampling results

The results of the monitoring, examination, testing, sampling activities carried out during the implementation of the AM programmes may indicate the actual occurrence and development of the assumed degradation mechanisms, and can also verify that the effects assumed in the SAMP have not yet occurred. If unexpected trends or the occurrence of previously not assumed ageing effects is experienced, a review of the AMPs must be carried out, with the possibility of developing and introducing new programmes as necessary.

02.4.4.3. Consideration of the most recent research activity results

The main R&D activities related to AM are summarised in Section 02.3.2.5.

An important element of the OAMP is the follow-up of the result of research activities, utilization of the results as necessary to maintain the effectiveness of the AM programmes. This is essential in cases where the properties related to the occurrence, development of an assumed degradation mechanisms contain significant uncertainties or for example do not cover the extended service life.

TBE304 [14] stipulates that the processing and evaluation of the technical-scientific results shall be carried out by the ageing management expert, who will make use of:

- Papers of domestic and international conferences concerning ageing management;
- R&D results, studies, analyses of organizations and companies working in the field of ageing management;
- Documents and reports of WANO, IAEA and EPRI.

R&D reports, working group documents of conferences and information in any other form deemed useful from the power plant's ageing management aspect are part of the ageing management knowledge base. In accordance with the provisions of TBE304 [14] elements and contact details of the knowledge base available on paper and/or electronically are recorded, maintained and stored by the ageing management experts.

The latest available research results are considered during the 5-year review of the mechanical DAMPs/SAMPs, the structural AM programmes, and the electrical and I&C AM programmes, as well as at the ageing management section of the PSR.

02.4.4.4. Consideration of legislative changes

Periodic amendments of the legislation concerning AM are considered during the 5-year reviews of the regulatory background of the OAMP, DAMPs, SAMPs, operative programmes and their regulatory procedures and instructions as well as at the ageing management section of the PSR.

02.4.4.5. Identification of further research activities

The main R&D activities related to AM are summarised in Section 02.3.2.5.

To follow the international trends and to address the uncertainties, inadequacies of own AM experience, the Ageing Management Section recently identified further research tasks to be carried out during the period of the service life extension, and with the involvement of the Structural Integrity Scientific Advisory Board defined the list of additional long-term R&D activities related to the ageing management of the main components. The list of activities supports the optimal utilization of R&D resources, summarizes and defines the specific topics, the expected results as well as ranks them according to their importance.

Tasks of high priority:

- Research and investigation activities related to the ageing management of the radiation swelling and stress corrosion cracking of the reactor vessel internals;
- Investigations related to the development of the AM of the dissimilar welds of the steam generators.

Non-priority, but important tasks:

- Assessment and preparation of the application of small specimen, and NDT methods to be used for trending and monitoring of given degradation mechanisms of the main components;
- Investigation of corrosion products and sludge on the primary and secondary side of the steam generator tubes to get more information about the fouling process and to evaluate the cleaning possibilities;
- Investigation of the ageing of welded cladding;
- Investigations of thermal ageing of cast stainless steels;
- Application of advanced numeric methods
  - for the determination of the probability of the brittle fracture of the reactor pressure vessels;
  - for the assessment of coolant-metal boundary processes related to the thermal fatigue analysis of the T- junctions.

02.4.4.6. Review of time limited ageing analyses

There were no original TLAAs available in sufficient details prior to the licensing of service life extension of Units 1-4. that could have been used simply, but according to current requirements when extending or modifying the TLAA through the reduction of the conservatism of the original analysis to a justified extent, or maintaining its validity through ageing management measures. Therefore, in the practice of Paks NPP, review of the TLAAs was essentially the identification and replacing the TLAAs. The TLAAs, the final scope of which is shown in table 02.4.4.6-1 were prepared during the preparation for the service life extension licensing of Units 1-4., between 2008 and 2016.

Table 02.4.4.6-1: List of TLAAs managed in the national practice

Number	Name of the TLAAs
1.	Low cycle fatigue analysis of ABOS 1-2 mechanical components
2.	Environmental qualification of electric, I&C components
3.	PTS analysis of the RPV
4.	Determination of p-T curves of the primary circuit
5.	Crack propagation analysis
6.	Thermal stratification analysis of ABOS 1-2 pipelines
7.	Extension of the scope of safety analyses corresponding to high energy line break
8.	High cycle fatigue analysis due to flow induced vibration of RPV internals

Number	Name of the TLAAs
9.	High cycle fatigue analysis due to flow induced vibration of steam generator internals
10.	Material property change analysis of RPV internals
11.	Fatigue analysis of hermetic penetrations
12.	Fatigue analysis of components belonging to hermetic lining (welds)
13.	Thermal embrittlement analysis of Safety Class 1-2 mechanical components
14.	Fatigue analysis of cranes performing safety functions
15.	Fatigue analysis of spent fuel pool cladding
16.	Material property change analysis of steam generator heat exchanger tubes
17.	Material property change analysis of heavy concrete structures with shielding function
18.	Analyses of increased pressure integrated leakage tests
19.	Analysis of corrosion wall thickness allowance
20.	Fatigue analysis of the main circulating pump flywheel
21.	Analysis of building settlement and its consequences
22.	Evaluation of B^{10} loss of spent fuel pool grids
23.	Analysis of the impacts of cracks embedded due to the weakening of the grain boundary heat affected zone of the cladding of the RPV
24.	Analysis of the change of material properties of the ceramic thermal insulation of the upper block
25.	Fatigue analysis of the spent fuel pool racks
26.	Fatigue analysis of the refuelling machine
27.	Fatigue analysis of hermetic compartment hatches
28.	Fatigue analysis of diesel generators
29.	High cycle thermal fatigue analysis of the main circulating pump flywheel and head
30	Lifetime limit analysis of the heat insulating steel structure of the reactor pit wall

Review of the TLAAs was also carried out in 2017 as part of the 10-year PSR; the validity of the TLAAs with some corrective measures was demonstrated. In the clear majority of the cases the TLAAs are still valid, because the conditions, baseline data taken into consideration during their preparation have not changed and according to our current knowledge the applied analysis methods meet the requirements.

02.4.4.7. Introduction of new information into the AMP

The introduction of unpredicted degradation mechanisms arising from own and/or industrial AM experiences as well as new requirements, suggestions, information related to the AMP introduced domestically and/or in a relevant regulatory environment to the ageing management programmes takes place during the regular 5-year AMP reviews, the 10-year PSR or if necessary on an ad hoc review basis of the AMP. The compilation of the annual AM review may also involve the introduction of new information where justified by discrepancies identified during the implementation of a particular SAMP.

02.4.4.8. Periodic evaluation of the effectiveness and evaluation of AM

In accordance with the procedures set forth in TBE304 [14] evaluation of the overall ageing management is carried out considering the current regulatory framework as well as the internal regulations of the Paks NPP. In accordance with Guideline 1.24 [20] issued by the HAEA summary of the AM evaluation must be included in the annual report. The evaluation is conducted by the AM Section, for which data is provided by the specialized fields. Details of the evaluation procedure are contained in TBE304 [14], according to the following main elements, features:

- Evaluation of priority mechanical equipment should be carried out individually, non-priority mechanical equipment may be grouped together;
- The implementation of the annual AMP tasks and processing of the results must be verified;
- The evaluation must take into account the previously obtained comments of the HAEA;
- In case of discrepancies identified during the annual evaluation, review, modification of the DAMP/SAMP or modification of the scope of AM/SAMP may be necessary;
- The adequacy of the OAMP is proven also by the HAEA's assessments.

Budapest Research Reactor

Management of audit and inspection results

The AMP is evaluated on an annual basis and every 10 years during the Periodic Safety Review. The availability and failure rates are also evaluation parameters. These numbers have not increased significantly in the last 12 years from the introduction of the AMP, except for the two exceptions.

Review of the programme takes place every 5 years or if such an event occurs that can be attributed to the ageing processes of similar reactors. Review can be initiated by the change of the regulatory environment, regulatory inspection and the regulatory resolution completing the 10-yearly conducted PSR.

Areas currently considered problematic:

- Technology obsolescence;
- Change or disappearance of contractors;
- Ageing of operators, lack of professionals.

The operator organization of the research reactor wishes to address these challenges by the introduction of new diagnostic and measurement technologies, and civil servant life carrier planning.

The basis for the evaluation of the AMP is the acceptance and compliance parameters. Beyond that, during the PSR the residual lifetime of the SSCs within the AMP shall be demonstrated. This estimated lifetime shall extend over the next 10 years interval. If the SSC is not probable to comply with this requirement, then a short or middle term replacement programme shall be developed.

Evaluation of operating experience

The operating, maintenance and ageing management organization of the BRR are not sharply separated from each other; a person usually acts at more areas. The not very complicated technological systems and the small staff make the operation and maintenance of the reactor transparent. Evaluation of the experiences takes place during the group leader meetings. The results appear in the regular reports (refuelling related reports, annual report, PSR report).

Evaluation of effects of modifications

The following modifications affected the area of ageing management:

	Modification	Effect
1	Fuel conversion to low enriched uranium	Non-measurable.
2	Replacement of make-up waster production system	Positive, better parameters of the treated water.
3	Replacement of main gate valve sealing	Positive, min. 10 years residual lifetime.
4	Replacement of secondary circuit filter	Positive, better filtered water parameters.
5	Wet to semi-dry modification of spent fuel storage	Positive, decreased corrosion rate.
6	Mechanical filter blanket was place on the top of the iodine filters	Positive, the carbon cartridge does not saturates with dust.

Consideration of amendments of legal requirements

Review of the ageing management activity takes place after the modification of the legal requirements. The review takes place extraordinarily, during the 5-yearly regular review or during the 10-yearly PSR.

Integration of new information into the AMP:

Beyond the reviews, the following improvement proposals were integrated into the AMP of the BRR:

- Gamma monitoring at the leakage detecting wells of the external spent fuel storage;
- Regular sampling and measurement of the liquid radioactive waste storage tank.

02.5 Licensee's experience of application of the overall AMP

Paks NPP

02.5.1. Summary assessment

The OAMP system of Paks NPP complies with the relevant national requirements and the international practices. The ageing management programmes of Paks NPP meet the requirements related to AM set out by the HAEA, WENRA and the IAEA, and follow the international recommendations aimed at improving the effectiveness of AM.

02.5.1.1. Examples of AM good practices

In the overall ageing management programme of the Paks NPP, the system for designing and operating the equipment and component specific AM programmes (SAMP), which in a documented form and specifically covers the range of components important to safety within the scope of AM, can be considered as a good practice. Every SAMP contains the 10 or 9 attribute AM programme elements used in relevant international practices (NUREG 1801 GALL [3], IGALL [9]). This provides a suitable framework for all the conditions, activities required for ageing management in the component group of a particular SAMP or particular equipment, and clearly assigns operative AM programmes to them.

An additional good practice is the AM knowledge base and the development and continuous maintenance of the role of the DAMP for updating own and international experiences, technical-scientific knowledge related to ageing management, and the utilization of the additional knowledge in the SAMPs.

02.5.1.2. Deficiencies of AM practices

No deficiencies have been identified during the current review.

Budapest Research Reactor

The programme has been revised twice: in 2007, after the modification of the Nuclear Safety Code and in 2009, after the modification of the government decree regulating the pressure test of the pressure vessels. The stipulations were taken into account in both cases. The comprehensive inspections of the authority did not conclude any significant deviation or improvement action in relation to the programme.
The organization operating the programme performs adequately, no change is justified.
Concerning the monitored parameters, the fuel wall temperature measurement was terminated in 2006, and the programme was extended with the waster sampling of the liquid radioactive waste tank.
The results of primary and secondary circuit waters samples are excellent. Partial replacement of the primary medium took place three times, while the secondary circuit received an ion exchanger filter.
The residual lifetime of the equipment is min. 10 years, but based on the recommendations of the manufacturer the batteries, chargers and inverters were replaced.
The buildings are aged. Modernization was launched, but the weakest point of the programme is the condition of the technological buildings and the unused structures.

02.6 Regulatory oversight process

According to the Act on Atomic Energy, the nuclear facilities in Hungary are under permanent oversight. The government decree containing the detailed requirements [A8] determines the frames for that. The HAEA verifies through its licensing activity that the requirements are met before the commencement of any nuclear safety related activity, including both those related to the life-cycle of the facilities and to any modifications. Regulatory inspection and review and assessment activity ensure that compliance with the requirements is maintained throughout the activity. The HAEA has authorization to launch an enforcement procedure if a non-compliance with the requirements is detected.

Regulatory oversight of ageing management is fully embedded in the scope of the above described regulatory oversight process. It is part of the licensing process as described in Section 02.1, the inspection programme performed by the HAEA including event investigations and comprehensive inspections and of the safety performance assessment of the facilities performed by the HAEA.

Ageing management appears as an individual topic of the 10-yearly due Periodic Safety Review. The purpose of the review is to assess the efficiency of the ageing management programme. In order for that, on the one hand the elements of the ageing management programme and the condition of the SSCs in the scope shall be described and evaluated and the related documentation shall be reviewed and analyzed, while on the other hand upon analyses the deficiencies of the programme shall be determined and the necessary improvement actions shall be specified. An important aspect of the review and determination of the actions is to evaluate them in the mirror of national and international experiences and state-of-the-art practices and recommendations. During the PSR the HAEA reviews the assessment results of the Licensees, performs inspections as necessary and authorized to determine further conditions, safety improvement actions.

In the regulatory inspection system, in addition to the event and revealing inspections associated with the licensing procedures, submissions and events and the planned regular inspections, the internal procedure [A19] of the HAEA concerning the comprehensive inspections contains ageing management as a designated area (number B12). During the planning of comprehensive inspections efforts should be made that the thorough inspection of the designated areas between two PSRs takes place at least twice. At the processes of the facility, the works of the organizational units, compliance with the requirements and practicality should be evaluated first of all from the aspects of safety and quality. During the inspection the following topics should be especially considered:

- procedures applied in the work processes and their suitability,
- determination and unambiguity of the interfaces between the specific work phases of the work processes,
- feed backs associated with the process and their utilization (active learning),
- compliance of the auxiliary processes of the inspected main process,
- safety level of the facility.

Assessment of ageing issues is part of the Licensees' regular and (reportable) event reports and evaluation of the ageing management activities. The HAEA reviews annually the report of the licensees on the evaluation of ageing management. The results are taken into account in the report prepared about safety performance assessment of the facilities concluded based on the regular reports, safety indicators and the licensing and inspection experiences.

Regulatory oversight of extension of design service life of the nuclear power plant should be mentioned separately. The design lifetime of the Paks NPP units expired between 2012 and 2017. According to the regulations, the licensing of it was a two-step process. In the first one, 4 years before the expiry of the design service life, the programme meant to outline the preparation for extension had to be submitted. The Licensee prepared one programme for all units. In the second step the actual licensing took place unit by unit. In line with the requirements the ageing management activity had to be assessed from two aspects. In the case of the passive, long-lived components a comprehensive review of the ageing management had to be performed, while in the case of active or replaceable components the efficiency of the AMP had to be demonstrated. The HAEA paid special attention during the review and inspection of implementation of the programme to confirm the compliance of the ageing management review and efficient operation of the AMP.

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

The HAEA reviewed the information submitted by the licensees of Paks NPP and Budapest Research Reactor and compared it to the information deriving from regular reports, licensing activities and inspections. In the case of Paks NPP the primary source of information originates from the regulatory oversight activities performed during the service life extension of the specific units, during which all aspects of ageing management of the facility was scrutinized. Ageing management activity of the Budapest Research Reactor was last examined and assessed in 2015 in the frame of a comprehensive inspection. Based on the information from these activities the HAEA agrees with the statements and conclusions taken by the Licensees in relation to Chapter 02, as follows.

The requirements on the ageing management programme of the nuclear facilities in Hungary are in agreement with the WENRA reference levels and the IAEA requirements and recommendations. There are detailed legal requirements to regulate the design and operational aspects of ageing and ageing management, its assessment during the PSR and regulatory measures are used in this are to provide appropriate oversight.

The HAEA, by systematically using its oversight measures, verifies the compliance of the overall ageing management programmes of the Licensees and its operation.

Based on the reports of the Licensees and the results of the oversight activity of the HAEA the overall ageing management programme of both Paks NPP and the Budapest Research Reactor meets the legal requirements. The programmes cover the ageing mechanisms, the means to identify and monitor them, the related limits and actions to implement if exceeding them, the quality assurance requirements and the feedback of national and international experience. The licensees are also prepared to identify the unforeseen ageing mechanisms.

03. Electrical cables

Paks NPP

03.1 Description of ageing management programmes for electrical cables

03.1.1 Scope of ageing management for electrical cables

Records of safety relevant cables can be found in the plant's cable database (ADRIA). For each entry, the database cable list contains the cable ID, the commodity group identifier for the environmental qualification, the start and terminal equipment, the type and length of the cable and includes information on the related safety feature, according to which:

- It is required for incident management (yes or no);
- It has safety function (Safety Related = SR), but is not subjected to harsh environment (or may be subject to harsh environment, but in that case performance of its function is not required, SR=1);
- It has safety function and it may be subject to harsh environment, in which its operation is required (SR=2);
- It performs safety function during severe accident management (SR=3).

Table 03.1.1-1 of the Annex contains the main information for the management of the commodity groups of electrical cables, which WENRA expects to be presented in the national report.

The cables of the neutron flux measurements are presented in a separate group, in the second table of the annex. The ageing management of these cables was evaluated similarly to the other cables during the compilation of the National Assessment Report.

03.1.1.1. Determination of the degradation mechanisms

The ageing management of the clear majority of safety important cables performed using environmental qualifications in accordance with the IAEA DS485 [2] Guideline. The determination of their degradation mechanisms is part of the environmental qualification. Degradation due to thermal effects and irradiation is taken into consideration through accelerated ageing in the qualification process.

In the case of some cables, cable groups as well as cable terminations (terminal bodies, terminal connections, bolt connections etc.) it is more suitable to manage the degradation mechanisms using ageing management programmes. In these cases, the degradation mechanisms were determined based on our own and relevant international experiences (NUREG-1801 [3], IAEA TECDOC-1402 and 1147 [30, 31], IAEA-EBP-LTI-22 [32]).

03.1.2 Ageing assessment of electrical cables

The cables recommended to be used as an example are the following:

- High voltage (>3 kV) cables subject to harsh environment (moisture, radiation, temperature);
- Medium voltage cables (380 V ≤ U ≤3kV) buried or covered (in trenches);
- Neutron flux instrumentation cables.

The environmental qualification programme is one of the accepted programmes for ageing management in line with international standards DS485 [2], IGALL AMP207 [33]. Based on this, the AM review of the sample cables is presented based on the environmental qualification programme of the respective cables as well as a few relevant V-SAMP in the next subsections.

03.1.2.1. Ageing mechanisms and acceptance criteria

Cables subject to environmental qualification

Environmental qualification tests of the cables were performed by accredited laboratories.

The acceptance criteria of the condition assessments of the cables during the artificial pre-ageing were the following:

- Insulation resistance measurement compliance criteria: based on the MSZ HD 60364-6:2007 [34] and MSZ IEC 13207:2000 [35] standards;
- Voltage test compliance criteria: based on the MSZ IEC 60502-1, 2 [36, 37] standards;
- Loss factor (tgδ) measurement compliance criteria: value specified by the cable manufacturer;
- Acceptance criteria of mechanical parameters (tensile strength, elongation at break): based on the MSZ EN 60811-1-1:1995 [38] standard.

If, accident condition testing was also required during the environmental qualification of a given commodity group, the acceptance criterion of the thermal/irradiation ageing effects of the respective cables was the expected functionality of the respective cable for the required duration.

Cables managed in the framework of V-SAMP-07

The AM programme addresses the following ageing effects on the components of cables:

- Insulation of conductors: oxidative ageing, ageing due to radiation, moisture, chemical ageing, maintenance load;
- Outer layer: oxidative ageing due to radiation.

Compliance criteria of the SAMP are the following:

- During discharge and return voltage measurements, no arcing or disruptive discharge takes place, the initial gradient of the discharge voltage is <100 V/sec;
- The average Shore D hardness on both end of the cable is <70. If a measurement exceeds the limit it must be noted in the records. The results are appropriate if the deviation of the measured values from the average is <10.

Cable terminations managed in the framework of V-SAMP-01, 02, 03, 04, 05, 06

The AM programmes address the following ageing effects on the cable termination:

- Metal parts, bolt connections: corrosion, loosening, maintenance load;
- Terminal body connectors: loosening, contamination, maintenance load.

SAMP acceptance criteria are the following:

- No traces of heat or corrosion;
- Connection parts are not movable;
- No apparent shear or fracture traces, no visible deformation.

03.1.2.2. Standards, guides and manufacturing documents

Standards, guides, manufacturing documents used in the environmental qualification of the sample cable commodity groups are the following:

- IEEE 323 [27];
- MSZ IEC 13207 [35] ;
- MSZ IEC 60502-2 [37];
- MSZ IEC 60502-1 [36];
- MSZ HD 60364-6 [34];
- MSZ EN 60811-1-1 [38];
- IEC 60230 [39];
- Guideline 3.15. [40];
- Documents originating from the cable manufacturers (supplier) describing the technical state, characteristics, stating the condition indicators and acceptance criteria in accordance with the requirements;
- NUREG-1801 [3];
- IAEA TECDOC-1402 [30];
- IAEA TECDOC-1147 [31];
- IAEA-EBP-LTI-22 [32].

03.1.2.3. R&D results used in the ageing management of cables

No R&D programmes were necessary for the environmental qualification of cable commodity groups.

In the case of the low voltage PVC insulated cables managed in V-SAMP-07, test methodology and acceptance criteria developed by the Budapest University of Technology and Economics (BME) were used.

03.1.2.4. Consideration of operational experience

In the environmental qualification of the sample cable commodity groups, application of the qualification standards constituted the application of industry operational experience.

Periodic review of internal and external operational experiences is part of the activity aimed at the sustainability of environmental qualification, regulated by TBE303: Procedure of Equipment Environmental Qualification and Maintaining of the Qualification [41]. In doing so the current status of the qualification and possible changes in the operational/accident environmental conditions are periodically reviewed; and, if necessary, requalification is performed as required.

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

03.1.3.1 Activities, methods, frequency of inspections

Cables subject to environmental qualification

The environmental qualification of the sample cable commodity groups in line with relevant international practice (IGALL AMP207 [33]) does not require condition monitoring, sampling and inspection activities during the qualified lifetime of the cables. However, periodic and ad hoc assessments of the environmental parameters are performed as part of qualification maintenance activities according to the TBE303 [41]. In addition, condition of some cables is monitored by OWTS (Oscillating Wave Test System) measurement methods as well, detailed in Chapter 3.2.3.

Environmental qualification of the cables and the prior definition of the qualification parameters and their monitoring take place during operation. In the time of the first PSR of the units the environmental parameters were defined as the enveloping values of the monitored during the operation real data. The environmental parameters were updated again based on measurements and calculations in the Final Safety Analyses Report of 2007, at this point, the possible impacts of the power uprate were also considered.

In 2017 during the PSR a review of the conditions of equipment environmental qualification was performed considering effects of modifications. It was concluded that the modifications did not affect the validity of equipment qualification.

Tests carried out on cables managed in the framework of V-SAMP-07

- Visual inspection of the mechanical condition and detection of colouration of the cable head insulation.
- Measurement of the initial gradient of discharge and return voltage. Measurements must be performed on 5 wires, in case of different colour wires; measurements must be performed on different coloured ones and in different geometric position. In case of the presence of a red wire it must also be investigated.
- Measurement of the Shore D hardness of the cable sheath at both ends of the cable at 5-5 points.

Measurements on the selected cable lines are to be performed every 4 campaigns.

Cable terminations managed in the framework of V-SAMP-01, 02, 03, 04, 05, 06

As part of the related maintenance programme, the AM programmes detect traces of heating and corrosion, and inspect the surface integrity of the components of the cable terminations at least once every 4 campaigns through visual inspection.

03.1.3.2. Consideration of inspection history

Consideration of the inspection history, possible trends, gradual degradation, consideration of unexpected degradation in the AM programmes of the sample cable commodity groups are as follows.

Cables subject to environmental qualification

The environmental qualification of the sample cable commodity groups in line with relevant international practice (IGALL AMP207 [33]) does not require actions related to the identification of inspection history, trends, gradual degradation or unexpected degradation during the qualified lifetime of the cables.

Cables managed in the framework of V-SAMP-07

- For the selected cables measurement of the change in the initial gradient of the discharge and return voltage must be monitored.
- Monitoring of changes in the cable sheath Shore D hardness of the selected cable line.

In case of deviations in the trends, inspections must be extended to previously not tested cable lines in the scope.

Cable terminations managed in the framework of V-SAMP-01, 02, 03, 04, 05

The frequency of damage occurrence of the cable termination components is monitored; the observed changes are evaluated based on the age of the examined elements.

03.1.4 Preventive and remedial actions for electrical cables

03.1.4.1. Preventive actions

Cables subject to environmental qualification

The environmental qualification programme of the sample cable commodity groups in line with relevant international practice (IGALL AMP207 [33]) does not require preventive actions during the qualified lifetime of the cables.

Cables managed in the framework of V-SAMP-07

As a preventive measure, the AM programme emphasises the need for caution during their maintenance to avoid cable loads.

Cable terminations managed in the framework of V-SAMP-01, 02, 03, 04, 05, 06

To prevent the ageing of cable terminations, the AM programmes require compliance with the required maintenance technology and surface cleaning.

03.1.4.2 Remedial actions

Cables subject to environmental qualification

The environmental qualification programme of the sample cable commodity groups in line with relevant international practice (IGALL AMP207 [33]) does not require corrective actions during the qualified lifetime of the cables. However, activities related to maintaining the qualification as contained in TBE303: Procedure of Equipment Environmental Qualification and Maintaining of the Qualification [41] can be considered as corrective actions. During these, it is evaluated whether any suspected deviations of the environmental parameters could have played a role in the possible malfunctions. Technical modifications necessary to restore the environmental parameters via operational intervention or to replace the equipment is carried out in accordance with the procedure contained in TBE206 [42].

Cables managed in the framework of V-SAMP-07

As a corrective action of the AM programme, first the technical assessment must be carried out, to identify possible unique impacts, factors independent of ageing. In cases where these can be excluded, an expert should be consulted, or in the case of deviations in the results of a given test, it is required that the tests should be extended to cover previously untested cable lines in the scope.

Cable terminations managed in the framework of V-SAMP-01, 02, 03, 04, 05, 06

As corrective actions of the cable terminations, the AM programme requires the removal of heat and corrosion stains and the replacement of defective connections as per the assembly instructions using appropriate materials.

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

03.2.1. Assessment of degradation mechanisms to be managed

Cables subject to environmental qualification

The environmental qualification programme of the sample cable commodity groups in line with relevant international practice (IGALL AMP207 [33]) does not require the assessment of the evolution of degradation mechanisms during the qualified lifetime of the cables, given that the evolution of the degradation mechanisms to be managed have already been taken into account by the standard EQ tests.

Cables managed in the framework of V-SAMP-07

Based on the experience of the development of the ageing management assessment methodology no significant ageing effect is expected on the installed PVC cables within the scope.

Cable terminations managed in the framework of V-SAMP-01, 02, 03, 04, 05, 06

Increase and trending of the number of damages and degree of damage found on the components of cable terminations is not characteristic.

03.2.2. Modifications in the programmes and their justification

Cables subject to environmental qualification

The current qualified condition of the sample cable commodity groups is in all cases satisfactory, no group is subject to time limited qualifications, all qualifications are valid until the end of the extended service life.

Modification of the environmental qualification programme of the sample cable commodity groups has so far not been necessary.

Cables managed in the framework of V-SAMP-07

Modification of the AM programme was not required.

Cable terminations managed in the framework of V-SAMP-01, 02, 03, 04, 05, 06

Modification of the AM programme was not required.

03.2.3. Conclusions of the licensee related to the ageing management of electrical cables

Cables subject to environmental qualification

The environmental qualification programmes of the sample cable commodity groups can be considered as ageing management of the specific cables, and it can be concluded that they comply with the relevant national requirements and international practices (IAEA DS485[2], IGALL AMP207 [33]).

The relevant programmes are (also) based on laboratory testing carried out in accordance with the standards prescribing the environmental qualification requirements for the cables of power plants; the methods used during the qualification and the acceptance criteria ensure the functionality of the respective cables during the cable's qualified lifetime.

Cables managed in the framework of V-SAMP-07

The structure, elements of the programme meet the requirements of the AMP, operational experience so far also supports the adequacy of the programme.

Cable terminations managed in the framework of V-SAMP-01, 02, 03, 04, 05, 06

The structure, elements of the programme meet the requirements of the AMP, operational experience so far also supports the adequacy of the programme.

In addition to the determination of adequacy of the cable AMPs the compliance-related experiences, challenges should be summed up according to the following important elements.

Middle voltage cables

For the qualification voltage test, insulation resistance measurement and loss factor measurement have been used to confirm the acceptability of the CG_SZAMK6 middle voltage cable type. Upon these conformity was verified.

After failure of a 6-kV cable in 2001 a decision was made to inspect the condition of middle voltage, plastic insulated cables belonging to the qualification group CG_SZAMK6 by OWTS (Oscillating Wave Test System) measurement method.

Tests were carried out in 2002 and 2003 in the scope of cables of equipment important to safety and production. The measurement results were applicable to identify initiation of the degradation of the cable penetrations, cable terminations and cable connectors and damaged condition of a cable section. Based on the OTWS failure survey repairs and replacements of cable terminations and cable extensions took place and further repair works were planned. OTWS cable measurement was introduced in every case where repair or installation of 6 kV cable had to be performed.

It might appear as a contradiction that the cables qualified for lifetime need to be replaced with new types at certain locations, but the real situation can be understood via a detailed explanation. The OWTS measurements revealed failure locations in the cable insulations and it is obvious that the cable is better if there are the less and weaker failure locations detected. However, there is not a justified maximum limit on the number of failures detected by this method, above which operational or incident failure of the cable can be expected. The number and extent of the failures detected by the OWTS method in the cable insulation is characteristic on an unfavourable feature of the insulation, and their existence is not good. For the lack of a limit therefore the plant decided to replace the old cables which had more failures.

The final result that the cables of important equipment are replaced to new ones (new types belong to the qualification groups CG_N2XS and CG_NTSCgEWöu) resulted in a better condition than the previous.

Instrumentation and control cables

In 1997, within the hermetic compartments, at several gate valves individual and cable route cable failures were found. Also, mechanical damages of patch cables and instrumentation and control cables were experienced. In the years after detection of the failures the plant performed field cable condition survey for the whole scope of hermetic compartment cables of each unit.

Activities on residual lifetime determination were commenced also at that time, thus the laboratory examinations required for the cable qualifications, thermal and dosimetry mapping of hermetic compartments for the qualification process were performed. A cable depot was established for the accelerated ageing of various cable type around the primary loop of Unit 2. One cable was removed each year after 1999 from the depot for the tests.

Cable replacements plans based on the cable condition survey and cable qualifications were developed until 2002. The cable replacements took place during the annual maintenance until 2010. During the activities the patch cables, individual cables of gate valves, plugs and intermediate boxes were replaced, connecting to which also the cable route cables were replaced depending on the revealed cable condition.

Replacements scheduled as described above or depending on the condition resulted in improved operational safety and decreased number of unplanned works.

Budapest Research Reactor

Scope of ageing management of electric cables

Among the cables managed by the Budapest Research Reactor there are only cables of or under 0.4 kV. The maintenance of primary side cables of the two transformers of 10/0.4 kV providing the electric supply of the facility are provided by the operators of the KFKI campus. The plastic sheathed cables of 10 kV are tested by the operator of the campus at nominal voltage per the Hungarian Standard MSZ 13207-2000 related to oil saturated, paper insulated cables. The test records are sent also to the research reactor for information. There are no cables of 0.4 kV buried or in trenches. The scope of current review exclusively covers for the cables of the nuclear measurement chains.

Ageing management assessment of electric cables

As a part of the ageing management programme, the insulation resistance measurements of the 7 nuclear measurement chains are performed after each core configuration change, that is 1-2 per year. These measurements serve sufficient information on the condition of the cables. If it becomes necessary based on the insulation resistance, the Licensee decides on the replacement of the cable, which means the replacement of the whole concerned measurement chain. This has taken place only once.

03.3 Regulator’s assessment and conclusions on ageing management of electrical cables

The Licensee of Paks NPP has described the ageing management processes and activities related to cables according to the specifications. The ageing management practice of cables complies with the national regulations and the international requirements and recommendations.

The specific ageing management programmes elaborated and applied by the Licensee of the Paks NPP for the electric areas perform well in the ageing management system. The strength of the system is the well-based, systematic structure, but attention should be paid to the degradation mechanisms of other areas on the condition of the cables..

In the case of the Budapest Research Reactor only the cables of the nuclear measurement chains belong to the scope of the current review. The insulation resistance of these are regularly measured and the whole chain can be replaced if necessary. The activity complies with the regulatory requirements.

04. Concealed pipework

Paks NPP

04.1 Description of ageing management programmes for concealed pipework

04.1.1 Scope of ageing management for concealed pipework

The scope of ageing management of concealed pipework includes those piping, ducts important to safety, which are not fully accessible or have pipe sections that are not fully accessible.

04.1.1.1. Selection, grouping methods and criteria for concealed pipework

The (partially or wholly) concealed piping from the scope of AM was identified. These were separated from the scope of operating medium and structural material SAMPs grouped according to table 02.3.1.3-1.

The specific ageing management programmes for concealed piping are listed in table 04.1.1.1-1. The table describes the typical material quality, operating environment and the way of limited access to the pipes of the pipeing group of the given SAMP.

Table 04.1.1.1-1: Specific ageing management programmes for concealed pipework

Concealed pipework SAMP ID	Pipework-group subject to SAMP
Z-SAMP-17	Spent Fuel Pool Cooling (TG) system, corrosion resistant steel, partially concealed piping encased in concrete
Z-SAMP-18	Partially concealed carbon steel piping in pipe tunnels, buried in soil, encased in concrete, covered in trenches used to transfer Danube water
Z-SAMP-19	Partially concealed corrosion resistant steel piping encased in concrete used to transfer presumably radioactive primary circuit water, other contaminated solutions
Z-SAMP-20	Partially concealed corrosion resistant steel piping encased in concrete used to transfer presumably non-radioactive primary circuit water, other contaminated solutions
Z-SAMP-21	Partially concealed piping in trenches, buried in soil, made of carbon steel, non-corrosion resistant steel used to transfer diesel oil.
Z-SAMP-22	Concealed piping (ducts) partially encased in concrete made of carbon steel, non-corrosion resistant steel used to transfer air
Z-SAMP -23	Concealed piping (ducts), (partially) encased in concrete made of corrosion resistant steel used to transfer air
Z-SAMP-24	Partially concealed piping in trenches, pipe tunnel, made of carbon steel, non-corrosion resistant steel used to transfer treated water

04.1.1.2. Determination of degradation mechanisms

The identification of degradation mechanisms and components of concealed pipework considered to be critical is based on the Annex of Guideline 4.12 [15]. In the case of external surface of concealed pipework, the guideline recommends the management of material loss due to ground water corrosion, whereas for internal surfaces it recommends the management of material loss due to general corrosion and biological corrosion.

Degradation mechanisms to be managed on the internal surface of the pipework are the same flow side degradation mechanisms that are determined for the given medium-group and the given pipe material-group.

Degradation mechanisms to be managed on the external surface differ from the typical external degradation mechanisms (material loss, formation of cracks related to general and local corrosion) defined for the given medium-group and the given pipe material-group if the external surface of the given section of pipes is in direct contact with the soil. In these cases, pipe material loss due to soil corrosion and possible material property change and degradation due to other causes of the external protective coating must also be added to the list of degradation mechanisms to be managed.

04.1.2 Ageing assessment for concealed pipework

As a sample, the following pipelines are shown:

- **Pipework containing radioactive effluents**, which are managed in the scope of Z-SAMP-17 and Z-SAMP-19;
- **Pipework that transfer fuel to emergency power supply**, which are managed in the scope of Z-SAMP-21;
- **Pipework providing essential service water providing cooling to SSCs important to safety**, which are managed in the scope of Z-SAMP-18.

04.1.2.1. Ageing mechanisms and acceptance criteria

The managed degradation mechanisms are the following:

Pipework containing radioactive effluents (Z-SAMP-17 and Z-SAMP-19)

- Local corrosion is assumed on the internal and external surfaces as well;
- Microbiologically influenced corrosion (MIC);
- Deposition;
- Loosening, wear of flanged joints.

The underlined mechanisms above are of utmost importance because their occurrence, the appearance of the ageing effect is typical in our own operational experience.

A typical example of this damage is the local corrosion phenomenon observed in the cooling system of the spent fuel pool. From the corrosion point of view, this system can be regarded as a complex, open system; several contributing factors and their interactions played a role in the observed corrosion phenomenon, first of all factors highlighted above. In the corrosion process the significance of individual effects varies and no single, well-defined factor causing the phenomenon could be identified.

In the vicinity of welded joints produced at site, a thermal oxide layer developed as a result of welding procedure violation. This oxide layer facilitated the attachment of microorganisms involved in microbiologically influenced corrosion. Bacteria produce adhesion proteins to adhere to the surface. The progression of this process was facilitated by the intermittent operation of the system, i.e. intermittent stagnant conditions.

Pipework that transfer fuel to emergency power supply (Z-SAMP-21)

- Local corrosion is assumed on the internal and external surfaces as well;
- Microbiologically influenced corrosion;
- Deposition;
- Loosening, wear of flanged joints;
- Soil corrosion.

There are no priority processes in the above list, because their occurrence, the appearance of ageing effect in our own operational experience has so far not been typical.

Pipework providing essential service water providing cooling to SSCs important to safety (Z-SAMP-18)

- Local corrosion is assumed on the internal and external surfaces as well;
- Microbiologically influenced corrosion;
- Deposition;

- Loosening, wear of flanged joints.
- Soil corrosion.

The underlined mechanisms above are of utmost importance because their occurrence, the appearance of ageing effect is typical in our own operational experience.

The occurrence of microbiological corrosion in the carbon steel pipework operated with Danube water has been noted already in the early stages of operation of the power plant, which in many cases has led to the leakages of the affected pipes. However, in the case of the sample component, through-wall faults of the service water pipework buried in soil due to microbiological corrosion has never been detected. Deposition in the service water pipework is also typical; however, their extent does not affect their ability to carry out their function according to the requirements.

Previously, microbiological corrosion in the corrosion-resistant pipework in boric acid operational environment was not considered presumable; however, the determination of the causes of corrosion damage in the pipework of the spent fuel pool in 2013 has made it clear that the occurrence of MIC cannot be excluded even in these systems.

The acceptance criteria for microbiological corrosion damages – which can also occur together with pitting – is the retention or recovery (replacement, repair) of wall thickness needed for the pipe strength.

If the deposits do not affect the function of the pipes, the ability to perform visual inspections during the periodic and internal condition inspections means the acceptance criteria for depositions.

04.1.2.2. Standards, guidelines and manufacturing documents

Key documents taken into account during the ageing management of concealed pipework and their application:

- Guideline 4.12 [15] to define the minimum initial scope of the degradation mechanisms/critical locations to be managed;
- IGALL [9] to define the degradation mechanisms, ageing management programme elements;
- MSZ 27003 [43], MSZ 27011 standards [12] to define the acceptability of detected indications;
- Available manufacturing documentation of the pipe material to compile the SAMP technical data.

04.1.2.3. R&D programmes related to the ageing management of concealed pipework

In the scope of ageing management of the sample pipework, R&D activities have been initiated aimed at determining the root causes of the corrosion damages of the cooling pipes of the spent fuel pool and the monitoring and management possibilities of MIC. Due to the complexity of the issue, specialists in the fields of corrosion, welding, materials science, operation and microbiology were involved to perform the tasks. The results of these studies and laboratory tests have confirmed the previous assumptions, i.e. microorganisms, that could survive and multiply in the given conditions (temperature, boric acid concentration), played a significant role in the corrosion damage of the spent fuel pool cooling system; furthermore, the studies established and validated microbiological methods and procedures of MIC monitoring that can also be utilized at Paks NPP.

04.1.2.4. Consideration of operational experience

In the case of the sample pipework the fact that the main elements of the IGALL [9] ageing management programme were taken into consideration during the development of the concealed pipework specific ageing management programmes is regarded as the application of external operational experience.

Our own internal operational experiences were also considered in that the possible occurrence of MIC damages like those found in the cooling pipes of the spent fuel pool were also taken into consideration in other SAMP of concealed pipework containing radioactive medium.

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

04.1.3.1. Activities, methods, inspection frequency

Monitoring, testing, sampling, inspection activities taken into considering in the sample pipework SAMPs:

Pipework containing radioactive effluents (Z-SAMP-17 and Z-SAMP-19)

- Visual inspection and/or other material testing every time when the concealed pipework become accessible for any reason;
- In-service pressure test in the frame of ISIP to detect possible loss of leak tightness every 10 years;
- In-service non-destructive inspection within the framework of the ISIP to detect corrosion damage and to inspect the progress of previously identified indications in the case of the selected section of the cooling pipework of the spent fuel pool encased in concrete in the scope of Z-SAMP-17, every 10 years;
- Visual inspection and/or other material testing at least every 10 years in areas where visual inspection, material testing can be carried out, in non-accessible areas in case the occurrence of defects is detected i.e. during in service pressure tests or single condition assessment;
- Detection of signs of possible leakage (spillage, wet soil, level decrease etc.) because of through wall defects;
- Detection of deposits, inspection of the permeability of pipe sections using special programmes;
- Corrosion tests (microbiological as well) done on a case-by-case basis to identify root cause of discovered degradations.

Pipework that transfer fuel to emergency power supply (Z-SAMP-21)

- Visual inspection and/or other material testing every time when concealed pipework become accessible for any reason;
- In-service pressure test to detect possible loss of leak tightness every 10 years as part of the condition review programme;
- Inspection of the integrity, required quality of the corrosion protective coating and the wall thickness reduction caused by corrosion of the carbon steel buried in soil through visual

inspection and/or other material testing during inspections of excavated reference pipe sections at least every 10 years;

- Visual inspection and/or other material testing at least every 10 years in areas where visual inspection, material testing can be carried out, in non-accessible areas if occurrence of defects is detected i.e. during in service pressure tests or single condition assessment;
- Detection of signs of possible leakage because of through wall defects during the in-service walkdowns;

Pipework providing essential service water for cooling of SSCs important to safety (Z-SAMP-18)

- Visual inspection and/or other material testing every time when the concealed pipework becomes accessible for any reason;
- In-service pressure test to detect possible loss of leak tightness during the ISIP every 10 years;
- Inspection of integrity, required quality of the corrosion protective coating and wall thickness reduction caused by corrosion of the carbon steel buried in soil through visual inspection and/or other material testing during inspections of excavated reference pipe sections at least every 10 years;
- Visual inspection and/or other material testing at least every 10 years in areas where visual inspection, material testing can be carried out, in non-accessible areas if occurrence of defects is detected i.e. during periodic pressure tests or single condition assessment;
- Detection of signs of possible leakage (spillage, wet soil, level decrease etc.) because of through wall defects during the in-service walkdowns;
- Corrosion tests (microbiological as well) done on a case-by-case basis to identify root cause of discovered degradations.

External contractors are also used for performing the in-service inspection and unique tests of the pipework buried in soil and for microbiological tests as well.

04.1.3.2. Consideration of inspection history

Monitoring of possible trends is realized through the trending of the change in wall thickness measured during the in-service condition review of concealed pipework.

The trend of possible increasing of the corrosion spots, pits, traces on the cooling pipework of the spent fuel pool detected during the visual inspection carried out every 3 years is also evaluated during the comparison of previous and new video recordings.

The corrosion damages experienced in the cooling pipes of the spent fuel pool in 2013 causing (also) through wall defects were the result of previously unexpected degradation mechanisms. Activities aimed at identifying the faults (destructive and non-destructive material testing, corrosion inspection, microbiological effect tests etc.) are parts of the ageing management programmes of the cooling pipes of the spent fuel pool.

04.1.4 Preventive and remedial actions for concealed pipework

04.1.4.1. Preventive actions

The typical preventive actions used in the specific ageing management programmes of the sample pipework are the following:

Preventing **microbiological corrosion damages**:

- Periodic cleaning of the interior surface of the pipes e.g. high pressure flushing in the case of cooling pipes of the spent fuel pool.

Preventing **soil corrosion damages**:

- Use of corrosion protective coatings to protect the external surface of the pipework buried in soil from soil corrosion.

04.1.4.2. Remedial actions

The typical corrective actions used in the specific ageing management programmes of the sample pipework are the following:

- Recovery of design condition through pipe repair, pipe section replacement using approved repair technologies;
- In case of soil corrosion damages, restoration of design condition of the pipework and the corrosion protective coating based on approved repair technologies.

When selecting the corrective actions to restore design condition, the safety, availability and economical aspects must be considered.

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

04.2.1. Assessment of degradation mechanisms to be managed

In relation to the degradation mechanisms to be managed identified in the scope of the sample pipework, it can be concluded based on the results of the repeated material inspections that development of the corrosion damage of the cooling pipework of the spent fuel pool, occurrence of new local corrosion damages and extension of previously unrepaired ones did not take place. Because of the periodic cleaning of the pipes, it is expected that in the future further damages will not occur.

04.2.2. Modifications in the programmes and their justification

To manage the corrosion damages experienced in the cooling pipes of the spent fuel pool in 2013 causing (also) through wall defects, modifications of the related ageing management programmes were required. On the one hand, it was necessary to carry out activities aimed at identifying the root cause of the previously not assumed degradations (material testing, corrosion and biological testing, evaluations); on the other hand, to develop and apply special non-destructive material testing techniques that made it possible to detect the corrosion indications of the concealed pipework.

The development and implementation of the periodic cleaning technology of the pipes was also a change to the ageing management programme.

In the first phase of organizing the piping into the SAMPs the specifics of the concealed pipeworks were covered by particular sub-groups within the framework of assigned SAMPs.

To ensure better management of the additional AM aspects of concealed pipeworks first, the TG-system's (cooling system of the spent fuel pool) partially concealed piping, where corrosion damages occured were separated to the Z-SÖKP-17, and then, as appropriate, all the concealed or partly concealed pipes were put into groups and SAMPs grouped to separate SAMP-s.

04.2.3. Licensee's conclusions about the ageing management of concealed pipework

It can be concluded about the specific ageing management programmes of the sample pipework, that

- They comply with IGALL [9] reference programmes for ageing management of concealed pipework;
- The AM programme developments, modifications introduced in recent years for the management of corrosion of the cooling pipes of the spent fuel pool serve as an example of satisfactory management of previously unexpected corrosion mechanisms according to requirements.

Operating experience with concealed pipeworks is summarized below:

1. Loss of tightness of stainless steel pipes of the spent fuel cooling system was experienced at many locations, which have not been expected previously (MIC). These pipes have 10-11 mm wall thickness, embedded in reinforced concrete and contain treated borated water of low temperature. Due to the recovery and repair works unit 3 was in shut down condition for several months and the related AMPs had to be improved including following factors:
   - In-service pressure test to timely detect the damages has not been sufficient; trends of the water level changes of the spent fuel pool indicates only the more significant leakages.
   - In-service visual and other non-destructive inspections of internal surface of the pipes have not been required, therefore significant developments were necessary to implement these inspections and incorporate them to the in-service inspection programme.
   - Examination methods, laboratory background necessary to determine the root cause of the corrosion damages had to be developed as well.
   - Introduction and application of tools and methods required for the replacement and local repair of the pipe sections embedded in reinforced concrete have meant also challenges.

2. Operating experience and condition assessments of the underground sections of the Essential Service Water (ESW) pipeworks demonstrated adequate condition of the pipes, through wall defects to the ESW pipelines have only occurred at the non-concealed sections. Nevertheless, to provide long term operability of the ESW pipeworks, regulatory licensing and preparation for implementation of repair of the pipeworks by plastic lining have been commenced.

Budapest Research Reactor

The following safety class pipelines of the technological systems of the research reactor belonging to the scope of ageing management run in the soil outside the buildings:

- TN01-TN04 suction lines of the ventilation systems;
- TZ pipeline to the liquid radioactive waste storage tanks;
- VA01; VA08 secondary pipelines.

The pipelines of the TN system run in the soil about a 30 m section. Visual inspection is performed by the opening of the so-called shaft C once every year. It is the suction side of the ventilation line, if it looses leaktightness, the sucked soil will appear in the in-built filters. The filters of the S1, S2 and S4 systems are inspected monthly. The estimated residual lifetime of the pipelines is minimum 10 years.

The material of the TZ pipelines is stainless steel. It is a double wall pipeline outside the buildings. Its visual inspection is not possible. If the line looses leaktightness the outer protection pipeline leads the water into the storage building that is visually inspected by the personnel at least once every week. Beyond that all the tanks of the system have water level measurement, so the quantity of released water and the liquid appearing in the storage tank can be well measured, and the balance unambiguously shows the appropriate integrity of the pipeline. The estimated residual lifetime of the pipelines is minimum 10 years.

The material of the secondary circuit VA01; VA08 pipelines is carbon steel. The pipelines exiting the reactor building run in a tunnel for about 40 meters then in the soil for about 80 meters. The condition of the secondary pipeline running in the tunnel is inspected visually at least once a week. In 2016 corrosion signs appeared at the sections that can be inspected visually. Wall thickness measurements were initiated then the pipeline was excavated in a 1.5 m section from the end of the pipe tunnel and 1.5-1.5 m from the two reversing shafts. The corrosion was significant. 30-37% corrosion loss was experienced of the pipeline wall of 7 mm nominal thickness. After full excavation holes and even worse corroded parts were found. During 2016-17 the replacement of the pipeline sections took place. The causes were identified as inadequate insulation technology, inadequate installation and de-icing of the surface roads with salt. The concerned pipelines were replaced, the use of salt was forbidden and also a weight limitation on vehicle transport was introduced within the area. The estimated residual lifetime of the pipelines is minimum 10 years.

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

Based on the detection of corrosion damages on the cooling pipelines of the spent fuel pool in 2013, the ageing management programme had to be modified. The performed inspections and research demonstrated that the new challenges required new inspection methods to be elaborated and introduced. The change of the ageing management programme made it necessary to change the approach (operating instructions, introduction of cleaning) and to recognize new relations of the reasons and consequences, such as those related to microbiological corrosion (MIC).

The concealed safety class pipelines of the research reactor covered by the ageing management programme, so the inspection, examination requirements are effective on them and they are managed accordingly.

05. Reactor pressure vessels

Paks Nuclear Power Plant

05.1 Description of ageing management programmes for RPVs

05.1.1 Scope of ageing management for RPVs

SAMP-001: Scope of the reactor pressure vessel specific ageing management programme:
Reactor pressure vessel:

- Safe end connecting to the main circulating pipes up to the first service weld;
- Leakage control of the head sealing to the weld before the outflow limiter;
- Safe end connected to the hydro accumulator pipelines up to the first service weld;
- Pipelines of in-vessel measurements from RPV up to the last weld before the flow preventers.

Vessel head (RPV head):

- a TF-CRDM (CRDM intermediate cooling circuit) inlet and outlet circuit pipes up to the main flange sealing of the upper block of the dismantable line in the reactor shaft space (the dismantable pipeline belongs to the CRDM intermediate cooling circuit);
- circuit pipe of the CRDM deaerator collector (TX) up to the first flange of the upper unit of the detachable line in the reactor shaft space (the detachable pipeline belongs to the system of controlled leakages);
- TX25 line of the gas removal up to the first flange of the upper unit of the detachable line in the reactor shaft space (the detachable pipeline belongs to the system of controlled leakages);
- seismic protection reinforcement.

Main flange sealing:

- individual components of the main flange sealing (stud bolts, nuts, washers, compression rings);
- leakage control piping of the sealing up to the weld after the outflow limiter within the reactor scope (connection to the leakage control system).

Table 05.1.1-1 is Table 1.3 of the RPV SAMP, which combines the degradation mechanisms to be considered and critical locations that are the most sensitive to the given ageing effect.

The figures in Annex 05.1.1-1 indicate the critical locations managed in SAMP-001 on the section drawing of the RPV except for the 14th critical location, the seismic protection reinforcements, which is not visible on the section drawing.

Table 05.1.1-1: Degradation mechanisms, critical locations taken into account in SAMP-001

	Critical component	Potential degradation mechanisms							
		Fatigue	Thermal ageing	Radiation embrittlement	Wear	Local corrosion (stress corrosion)	General corrosion	Boric acid corrosion	Loosening
1	RPV NA 500 nozzle	+	**+****					+ (outer surface)	
2	Cylindrical part (long belt) opposite to the core	+*	**+****	+				+ (outer surface)	
3	RPV 5/6 weld	+*	**+****	+				+ (outer surface)	
4	RPV 8,9/10 weld	+*	**+****					+ (outer surface)	
5	RPV vessel bottom	+*	**+****					+ (outer surface)	
6	Reactor pressure vessel main flange, M140 bolted joint with sealing and leakage control holes	+			+	+	+	+	+
7	Vessel head welds of the CRDM nozzles connecting to the internal surface of the dome	+				+			
8	Welds of the vessel head HE-FM penetrations connecting to the inner and outer surface of the dome	+				+			
9	Inner surface of the vessel head dome around the nozzles	+				+			
10	Outer surface of the vessel head dome around the nozzles						+	+	
11	Vessel head HE-FM nozzle M36 bolt connections and their surroundings	+			+	+	+	+	+
12	Vessel head CRDM nozzles M36 bolt connections and their surroundings with leakage control holes	+			+	+	+	+	+
13	Vessel head CRDM nozzles with inner tubes	+				+	+	+	
14	Seismic protection reinforcement of directly connected fixing/support structures						+	+	
15	Upper fixing weld of the liner tube of HE-FM nozzles	+							
16	Upper fixing weld of the liner tube of CRDM nozzles	+				+			

*The occurrence of fatigue cracks to be managed as indicated in Guideline 4.12 can be excluded during 50+10 service years based on the CUF<0.4 results of the fatigue analysis of the RPV, therefore the definition and inclusion of the AM programme elements for the management of fatigue in the given locations are not necessary.

**Management of the effects of thermal embrittlement indicated as to be managed in Guideline 4.12.: According to the results of the thermal embrittlement TLAA of the Safety Class 1-2 mechanical components management of the thermal ageing effects in the given locations is not needed therefore the definition and inclusion of the AM programme elements for the management of thermal ageing in the given locations are not necessary.

05.1.1.1. Method and criteria for determining the scope of the AM of reactor pressure vessels

The definition of the degradation mechanisms, ageing effects, critical components to be managed in the RPV SAMP have been performed based on the scope of ageing management required by the HAEA, contained in the Annex of Guideline 4.12 [15] that complies with the best international practice.

05.1.1.2. Determination of degradation mechanisms

Possible additional critical components/degradation mechanisms, expected ageing effect, own and VVER/PWR operational experiences not covered in Section 05.1.1.1, are added/can be added to the scope of RPV SAMP through the periodic or ad hoc SAMP reviews considering the latest IAEA, NRC, EPRI documents presenting relevant international AM practice.

05.1.2 Ageing assessment of RPVs

The following subsections were compiled based on the ageing management practice of recommended sample RPV components, such as:

- Base metal, cladding and welds of the steel vessel;
- Vessel head and the bottom including penetrations;
- Inlet and outlet nozzles.

05.1.2.1. Ageing mechanisms and acceptance criteria

Managed degradation mechanisms of the sample components considered in the RPV SAMP.

Base metal, cladding and welds

- Fatigue, including environmental effects at the inner surfaces in operational contact with the coolant;
- <u>Radiation embrittlement;</u>
- Boric acid corrosion (outer surface).

The underlined one from the degradation mechanisms above is of utmost importance because its occurrence, the effect of ageing has been experienced in our own operational practice as well.

Radiation embrittlement of the vessel wall is an occurring degradation mechanism causing continuous damage, characteristic of all VVER/PWR vessels, and based on current knowledge, in the scope of the sample components, this is the only degradation mechanism that can limit the permissible service life of the RPV.

Acceptance criteria related to radiation embrittlement:

Based on Volume 3 of Nuclear Safety Regulations that form annexes of the 118/2011 Government Decree [44], as well as Guideline 3.18 [23] issued by HAEA, the permissible ductile to brittle transition temperature of the given component determined using Pressurized Thermal Shock (PTS) analysis must be higher than the component’s actual critical temperature of brittleness including the specific safety margin during the service life of the vessel.

The RPV SAMP indicates the permissible ductile to brittle transition temperature determined using PTS calculations of the specific critical component, i.e. the cylindrical part belt line and circumferential weld located closest to the active core. SAMP-001 also specifies the vessel/component specific correlations required to determine the critical temperature of brittleness of the most critical components based on the actual fast neutron fluence; which were defined during the PTS calculations using the results of the RPV surveillance programme.

Vessel head and the lower dome including penetrations

- Fatigue, including environmental effects at the inner surfaces in contact with the primary coolant in the operation;
- Local corrosion mechanisms (including stress corrosion);
- Boric acid corrosion (outer surface).

The underlined mechanisms above are of utmost importance because their occurrences have been experienced in our own operational practice in accordance with the following:

- Cracking due to **fatigue** of the upper welds of the control rod nozzles liner tubes (sleeves) ensuring corrosion protection have repeatedly occurred, it is a known defect in the practice of Units 1-4 of Paks NPP and other VVER-440 vessels.
- **Boric acid** damage of the control rod nozzle, because of leakage of coolant at the flanged joints of the control rod nozzle, has occurred at the Paks NPP and other VVER-440 units.

Acceptance criteria of the fatigue:

- CUF>0.4 <1.0, this is the limit for which the location is considered to be critical from the aspect of fatigue; and management of the effects of fatigue at the given component must be required in the given SAMP;
- For CUF>1.0, operation can be continued using the analyses according to ASME BPVC XI [11] Appendix L (MSZ27011 [12] Annex L) contained in Guideline 3.25 [24] and additional monitoring of the given location.

Acceptance criteria of boric acid corrosion:

The permissibility of material losses due to boric acid corrosion must be evaluated based on the TLAS. The permissibility of the detected damage possibly exceeding the criteria indicated in the TLAS without reparation or replacement must be evaluated based on strength calculations.

Inlet and outlet nozzles

- Fatigue, including environmental effects at the inner surfaces in contact with the primary coolant in the operation;
- Boric acid corrosion (outer surface).

Of the listed processes above, the underlined fatigue can be considered more significant. This is because there are locations on the sample components where the results of fatigue calculations were CUF>0.4. This CUF limit used in the national practice should be considered critical from a fatigue aspect and must be managed in the SAMP of the given component.

05.1.2.2. Standards, guidelines, manufacturing documents

Documents taken into consideration in the ageing management of RPVs and their main uses:

- Volume 3 of the NSC [44] to determine the permissibility criteria for radiation embrittlement;
- Guideline 4.12 [15] to define the minimum list of degradation mechanisms/critical locations to be managed;
- Guideline 3.18 [23] to define the acceptance criteria for radiation embrittlement;
- Guideline 3.25 [24] to define the acceptance criteria of fatigue;
- MSZ 27003 standard [43] to define the acceptance criteria for fatigue;
- ASME BPVC XI standard [11] to define permissibility criteria of defects detected during material testing;
- ASME BPVC III standard [45] to define the acceptance criteria for fatigue;
- MSZ 27011 standard [12] to define permissibility criteria of defects detected during material testing;
- RPV and upper block passports, reactor equipment technical specification and operating instructions, RPV – material specification (Units 1-4), RPV – Units 1-4 strength calculations part 1 strength calculation of welded vessel, RPV – Unit 1-4 strength calculations part 4 PTS calculation of welded vessel, Upper block – strength calculations (Units 1, 4), to determine AM basic data, material qualities, material properties, construction features;
- Quality control programme of RPV and reactor head, to determine AM basic data, material qualities, material properties;
- Manufacturer drawings to determine AM basic data, construction features.

05.1.2.3. R&D programmes used in the ageing management of RPVs

A large portion of the recently finished and utilized major R&D topics supported the service life extension and ageing management. There are/were priority R&D projects related to the justification of the possibility of service life extension and management of occurring degradation mechanisms, damages of the sample components in accordance with the following:

- **Preparation, implementation of the cavity dosimetry measurements of the RPV:** To evaluate the embrittlement of the RPV wall the fast neutron fluence of the wall can be obtained using validated neutron physics calculations and/or measurements. Neutron monitors (activation detectors) placed on the outer surface of the vessel wall can also be used to evaluate the effect of the current fuel configuration, and the effect of its change on the fast neutron dose of the vessel wall. Validation of the reactor physics calculations were carried out during the new national surveillance programme based on the evaluated results of the neutron monitors irradiated in surveillance position. Measurements results of the neutron detectors placed on the outer surface of the vessel wall made it possible to evaluate/validate the results of the neutron transport calculations also on the outer surface of the RPV wall. The cavity dosimetry programme of Unit 2 was launched at the same time as the 'New National Surveillance Programme 2' (NNEP2) in 2012. The stand containing the activation neutron monitors were placed on the outer surface of the vessel, 'opposite' to the NNEP2 four-year irradiation chain. 'Monitoring packages' used to determine the neutron spectrum could be placed in four different locations in the stand: in the centre line of the zone, at the height of the 5/6 weld, and at the azimuth minimum and maximum. Verification of the two points confirmed that the calculation model determining the attenuation of neutron flux within the wall was correct.

- **Development of reliable repair technologies to manage the failure of the upper weld of the control rod nozzle inner tube:** As part of the research activities, new applicable welding material, welding technologies, inner tube materials were investigated. The results of the developments have created the conditions for implementing proper quality welds in the new repairs. Welding technological changes form a deliberate and coherent system and based on the result of all the inspections it can be stated that the repaired welds are of better quality, are more resistant to loads than the factory weld, and further damage of the repaired welds is less likely.

05.1.2.4. Consideration of operational experience

Significant modification of the AM programmes of the sample components have taken place based on internal and external operating experience in the following cases:

- In the case of PWR vessels, or e.g. in the Finnish nuclear power plant operating VVER-440/213 units, it was a proven operational experience to apply specific core configurations aimed at reducing the fast neutron fluence on the vessel wall. Based on this practice, to reduce the embrittlement of the vessel wall components opposite to the core, following the first few years of operation such fuel was introduced in Units 1-4 of the Paks NPP that resulted in lower fast neutron fluence on the wall of the vessel.
- There have also been external operating experiences for using Dummy elements (Loviisa) to reduce the fast neutron irradiation of the vessel wall, but the SKODA manufactured Paks NPP Unit 1-4 vessels based on the lower Cu and P contents are less sensitive to embrittlement, there was no need to introduce Dummy elements.
- There have been some cases of internal and external VVER operational experience where the leakage of the vessel head flanged joints of the control rod nozzle resulted in the leakage of the primary circuit coolant, causing boric acid corrosion material loss of the nozzles. To prevent further occurrence of these damages, a new YC system was installed to detect possible leakages of the joints of the control rod nozzle and have introduced on-site inspections of the vessel head during operation.

05.1.3 Monitoring, testing, sampling and inspection activities for RPVs

05.1.3.1. Activities, methods, inspection frequency

Monitoring, testing, sampling, inspection activities on the RPV sample components:

- **Implementation framework programmes of the in service non-destructive testing (NDT)**
  - ‘Reactor vessel and sealing units’ Material testing framework programme (KA-01_C15),
  - and the ‘Upper block’ Material testing framework programme (KA-02_C15).

Periodic material testing of the RPV and the upper block is carried out using the methods required for vessel components in the inspection table of the programmes using the inspection technologies indicated in Table 05.1.3.1-1. In addition to the visual inspection of the internal and external surface using TV cameras and the eddy current tests (ET) of the surface layer of the cladding, the pressure retaining wall of the RPV, the welded joints are inspected using ultrasonic testing (UT). In addition to inspections from the outer surface of the vessel, since the first main outage involving the complete removal of the zone of Unit 1. in 1987, the vessels have also been inspected from the inner surface that also demonstrated the commitment to safety.

Table 05.1.3.1-1: Non-destructive examinations used in RPV inspection

Topic of inspection technology
Eddy current testing carried out from the inner surface of the RPV
Ultrasonic testing carried out from the inner surface of the RPV
Surface examination of austenite and ferrite-perlite materials using liquid penetration test (PT)
Ultrasonic examination to detect water between the base metal of CRDM nozzles and the liner tube
Ultrasonic examination of the base metal and welded joints of ferrite-perlite steel components in the case of components with a diameter larger than 465 mm and wall thickness more than 15 mm
Manual ultrasonic testing of cladding adhesion
External ultrasonic inspection of the circular weld of RPV cylindrical part and the base metal of the belt line region
External ultrasonic inspection of the dissimilar metal weld of the DN 500 nozzle
External ultrasonic inspection of the dissimilar metal weld of the DN 250 nozzle
External ultrasonic inspection of the inner bend of the DN 250 nozzle
External ultrasonic inspection of the reactor pressure vessel nozzle belt circular weld
Visual inspection of components, pipelines

In-service NDT was carried out every 4 years until 2010, since then every 8 years; following the transition to a 15-month operating cycle, they will be conducted every 10 years.

The acceptability of deviations revealed during inspection is evaluated based on the acceptance criteria contained in the TLAS [17].

In the case of flaw sizes exceeding TLAS [17] level II the acceptability (fitness to operate further) must be supported by strength and fracture mechanics analysis. On the basis of the analysis the rules of periodic and ad hoc inspections and the date of the following test shall be determined. If the fitness to operate is not warranted repair/replacement is necessary, and subsequent repeated test to verify the suitability of the equipment.

Part of the periodic material testing, inspections performed from the inside of the vessel are carried out by contractors. The applied non-destructive testing systems are qualified according to the ENIQ methodology.

– **As part of the ISI programme of the RPV in-service pressure tests** are carried out during the ISIP of the reactors.

Currently the implementation frequency of the ISIP is 10 years. The temperature of the pressure test was determined by developing the p-T curves of the reactor cooling circuit. The method of the pressure tests and the acceptance criteria are contained in FEL008_VU06 [46] as well as the pressure test plan of the ISIP of the RPV. According to the acceptance criteria of the pressure test of the RPV, loss of pressure is not permitted.

- **RPV surveillance programme**

For the radiation embrittlement monitoring of the RPV structural materials, the reactors of Paks NPP were originally equipped with a set of six Loviisa type specimens. These sets contained tensile specimen, Charpy V-notch impact specimens and, for fracture mechanical testing, precracked static tensile specimens, fabricated from the base metal of the vessel, the material of the welded joints and the heat affected zone of the weld. The specimens were placed in hermetic capsules; 19 or 20 capsules represent one chain and two chains of capsules represent one set of specimens. There are 18 tensile, 36 impact and 36 fracture mechanic specimens (distributed equally from the previously mentioned three material types) in one set. Two sets, in addition to the above-mentioned specimens, contained specimens for monitoring thermal ageing (i.e. there were capsules above the core level as well). Certain capsules also contained indicators for measuring neutron flux and temperature. A higher number of set of specimens than previously mentioned was also delivered to determine the initial state (reference level).

The ductile-to-brittle transitional temperature of the structural material, i.e. the T_k critical embrittlement temperature was determined in the laboratory of the plant using Charpy impact specimens. The transition temperature curves were drawn using the impact energy, lateral expansion and ductile-to-brittle fracture surface ratio. The evaluation criteria of the critical embrittlement temperature in the given cases were: 41 J impact energy, 0.89 mm lateral expansion and 50 % ductile-brittle fracture surface ratio. The value of the upper shelf energy level was also determined. For each degradation parameter – as it is usual for Russian designed reactor – fast neutron fluence of E > 0,5MeV was used.

The value of the lead factors (LF) determined for the positions of the surveillance specimens at the locations of maximum flux was LF~12, therefore the plant, after the approval of the HAEA, scheduled the removal of the specimen sets after the first four operating campaigns of the reactors, because there was no practical significance to have an irradiation period longer than five years. Trend curves could be plotted based on the shift of the critical embrittlement temperature ΔT_k and taking into consideration the neutron fluence calculations. The results of the trend curves were used to position the fracture toughness reference curves of the structural materials K_{Ic}, and then the maximum allowed critical temperature of brittleness value was determined from the $K_I \leq K_{Ic}$ postulated crack stability criterion. The allowable service lifetimes T_k^{meg} for each reactor pressure vessel could be calculated based on the values and the vessel-specific critical temperature of brittleness**.** These calculations were performed by the HAS KFKI Atomic Energy Research Institute (now Centre of Energy Research) contracted by the plant. An independent expert body was contracted by the NPP for the complex verification of the reactor pressure vessel surveillance program (test, measurements, calculations, and evaluations).

In order to eliminate the weaknesses of the original surveillance programme and the resulting uncertainties (e.g. high lead factor, uncertainty of the position of the neutron flux monitors, inappropriate temperature inspection process) the plan developed and launched an additional surveillance programme in the beginning of the 1990s. This used specimens from the archive (restored Charpy) and reference material; the neutron flux monitors were placed in controlled positions and suitable temperature indicators were used. The new ('national') surveillance specimen sets were placed in the vacant trenches at the height of the constant flux and were replaced every three years. The results, on the one hand confirmed the results of the original surveillance programme, and on the other hand through the re-evaluation of the neutron fluence values, significantly reduced the scattering of the fluence values.

Since this supplementary surveillance programme was planned to last for 30 years of operation, and the service life extension obviously long-term surveillance of the reactor pressure vessel, the NPP ordered a further supplementary surveillance program to be developed. This programme takes into account the requirements of the ASTM E 185 (Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels), enabling the monitoring of the effect of neutron radiation during the entire lifetime of the reactor, furthermore it approaches the lead factor value to the expected range of the value in the above mentioned standard. The specimens of the supplementary surveillance programme are contained in three-three chains, which will be removed and inspected after irradiation of 4, 8 and 16 operational cycles. The programme has been started, its continuous evaluation will be performed. A new element of the programme is that it contains specimens made of the cladding material as well. The specimens are essentially the inlets of the static fracture mechanic and impact specimens with the appropriate notches, from which the complete specimen (specimen reconstruction) are formed with special welding following their removal from the vessel. In case of the cladding, miniature tensile specimens are also used.

Preparation and implementation of the surveillance programme are partly done with the involvement of external contractors.

- **Possible leakages of the primary circuit coolant** in the framework of the **RPV head surveillance programme**

are detected on the one hand through the operation of YC leakage detection system. In accordance with the 1-4PR49 operating instruction, the leakage of the flanged joints of the RPV is detected by the pressure signal in the space between the double sealing rings,

on the other hand, are detected during the 'ASV-1/2015- inspection of the reactor upper block' periodic walkdowns of the RPV head leakage surveillance programme.

The acceptance criteria of the monitoring, surveillance is that leakage is not permitted.

- **Cycle number monitoring programme**

The permissible values of the given cyclic loads of the cycle menu taken into account during the strength, fatigue calculations of the RPV taking into consideration the results of the performed fatigue analyses, are contained in Section 5.2.1.1 of the OLC [25] and its supplementary Annex 5.2.1.1.1. Recording of the RPV relevant items of the cycle menu is carried out continuously on the basis of the instruction of TBE301 Condition analysis and feedback procedure - TBE301_VU02: "Operational surveillance, plant analysis: Recording of operational surveillance data generated during the operation and maintenance of the units and performing operational analysis tasks". The acceptance criteria of the utilization of the cycle numbers, of the load run and the load properties as well are based on the load catalogue taken into account in the strength and fatigue calculations of the RPV.

05.1.3.2. Consideration of inspection history

Among the degradation mechanisms of the RPV components, trending has a role in the evaluation of irradiation embrittlement of the vessel wall components around the active core and in the fatigue analysis of the given components. The effects of the other degradation mechanisms managed in the RPV SAMP is not significant, so trending is not necessary.

The trend of the embrittlement because of fast neutron irradiation of the vessel wall components around the core is indicated by the increase of the critical embrittlement temperature (T_k) of the components. The trend curves of the components around the core of the vessels of Unit 1-4 defined based on the RPV surveillance programme and chemical composition are shown in Figures 05.1.3.2-1 and 05-1.3.2-2.

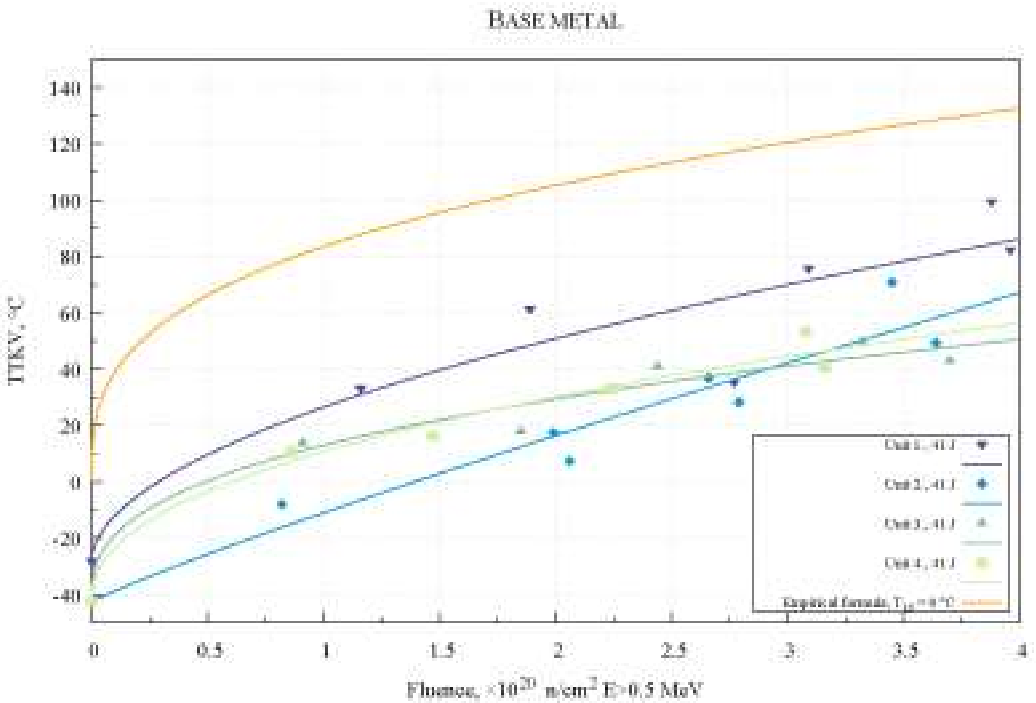

Figure 05.1.3.2.1-1: Tk trend curves of the base metal of the long belt of the Unit 1-4 vessels next to the core determined based on the RPV surveillance programme and chemical composition [47]

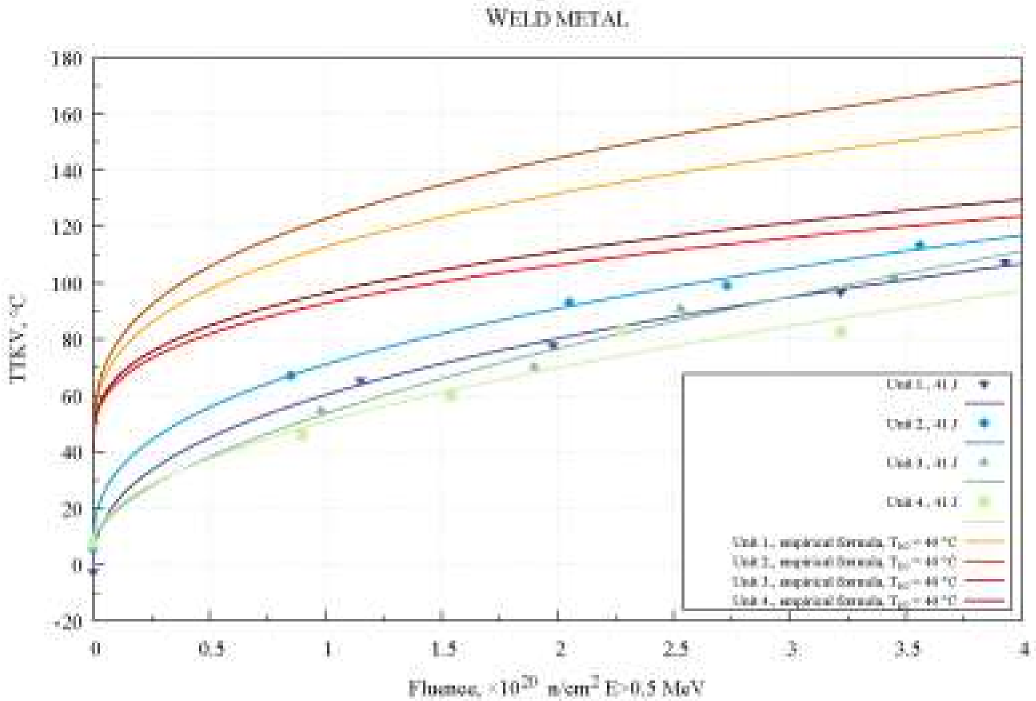

Figure 05.1.3.2.1-2: Tk trend curves of the circumferential weld of the vessel next to the core determined based on the RPV surveillance programme and chemical composition [47]

In the case of vessel components affected by **fatigue**, the gradual degradation, the load cycle number decrease trending taken into account in the component fatigue calculations are evaluated as part of the cycle number monitoring.

As an unexpected degradation mechanism, the effects, failures because of the CRDM nozzle inner tube fixing weld fatigue cracking can be considered. The identification of their occurrence is ensured by the since then introduced, periodic, cyclic inspections carried out by campaign as follows:

- UT testing carried out from the outer surface of the nozzle to detect the occurrence of water flowing into the space between the inner tube and the nozzle through the cracked fixing weld;
- Controlling the size of the gap between the control rod outer tube and the inner tube of the nozzle in order to find signs of buckling of the inner tube due to the pressure changes of the water in the space between the inner tube and the nozzle. The gap sizes between the control rod outer tube and the inner tube of the nozzle of the 37 CRDM nozzles are determined by plate gauges following dismantling of the CRDMs.

05.1.4 Preventive and remedial actions for RPVs

05.1.4.1. Preventive actions

The main measures, actions, procedures aimed at the prevention, reduction of the given degradation mechanism of the sample component taken into account in the RPV SAMP are as follows:

Preventing the initiation and propagation of cracks due to **fatigue**:

- Comprehensive consideration using sufficiently conservative values of the possible loads during the design fatigue calculations;
- Prevention from exceeding the cycle values determined during the strength calculations (TBE301-VU02).

Preventing material loss on the outer surface of the vessel due to **boric acid** corrosion:

- Timely detection of possible leakage of primary circuit coolant on the outer carbon steel surface of the RPV head (1-4PR49, ASV-1/2015).

Preventing material loss of the inner surface of the vessel due to **local corrosion**, preventing initiation and propagation of **stress corrosion cracks:**

- Maintaining corrosion-mitigating water regime parameters (01-02VE06: Primary and secondary circuit water norms).

Reducing the rate of **embrittlement due to neutron irradiation** of the components around the core:

- Reduction of the fast neutron fluence on the vessel wall components and its verification through calculation, analysis and inspection of the effects of utilization of various core configurations (see Section 05.1.2.4).

05.1.4.2. Remedial Actions

The typical corrective actions, activities, procedures in the case of degradation exceeding the acceptance criteria of the sample components taken into account in the RPV SAMP are the following:

- The application of unique, documented and authorized repair technologies for the repair of local forms of degradation (loss of material, cracks);
- Management of through wall cracking of the fixing welds of the control rod nozzle inner tubes by the replacement of the inner tubes, using new enhanced welding technologies, by application of authorized repair technologies;
- Elimination of possible boric acid leakage locations, restoration of the possibly damaged elements to their design conditions.

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

05.2.1. Assessment of degradation mechanisms to be managed

The main characteristics of the development of the degradation mechanisms identified to be managed are as follows:

Fatigue as a degradation mechanism has only occurred in the case of cracking of the fixing welds of the CRDM nozzle inner tubes. It is expected that due to the improved repair technologies used, further fatigue related through cracking will not occur at the already repaired nozzles.

Material loss of the RPV head due to boric acid corrosion has only occurred during the first phase of operation, since then the introduced and supplemented leakage detection activities proved to be effective.

The **local corrosion** mechanisms managed in SAMP-001 have not occurred so far in the practice of the Paks NPP in the scope of the sample components.

The rate of radiation embrittlement of the vessel components around the core presumed by the designer, and the actual or expected rate for the base metal of the long belt of the Unit 1-4 vessels next to the core is shown in Figure 05.1.3.2.1-1, for the circumferential weld of the vessel next to the core is shown in Figure 05.1.3.2.1-2. The figures show that the trend curves of the critical brittleness temperature determined based on the chemical composition of the components, calculated according to empirical formula, are more conservative than the expected T_k trend curves defined in the RPV surveillance programme.

05.2.2. Modifications in the programmes and their justification

Main modifications and justification of the ageing management programmes of the sample components taking into account the internal and external operational experience were described in Section 05.1.2.4.

The overview of the major changes in the RPV surveillance programme and their justification were detailed in Section 05.1.3.1.

The reduction of embrittlement of the components of the vessel wall opposite to the core was realised in the first few years of operation, when the applied core configuration reduced the fast neutron load to the vessel wall. This is visible on the example of Unit 1, which is shown in Figure 05.2.2-1.

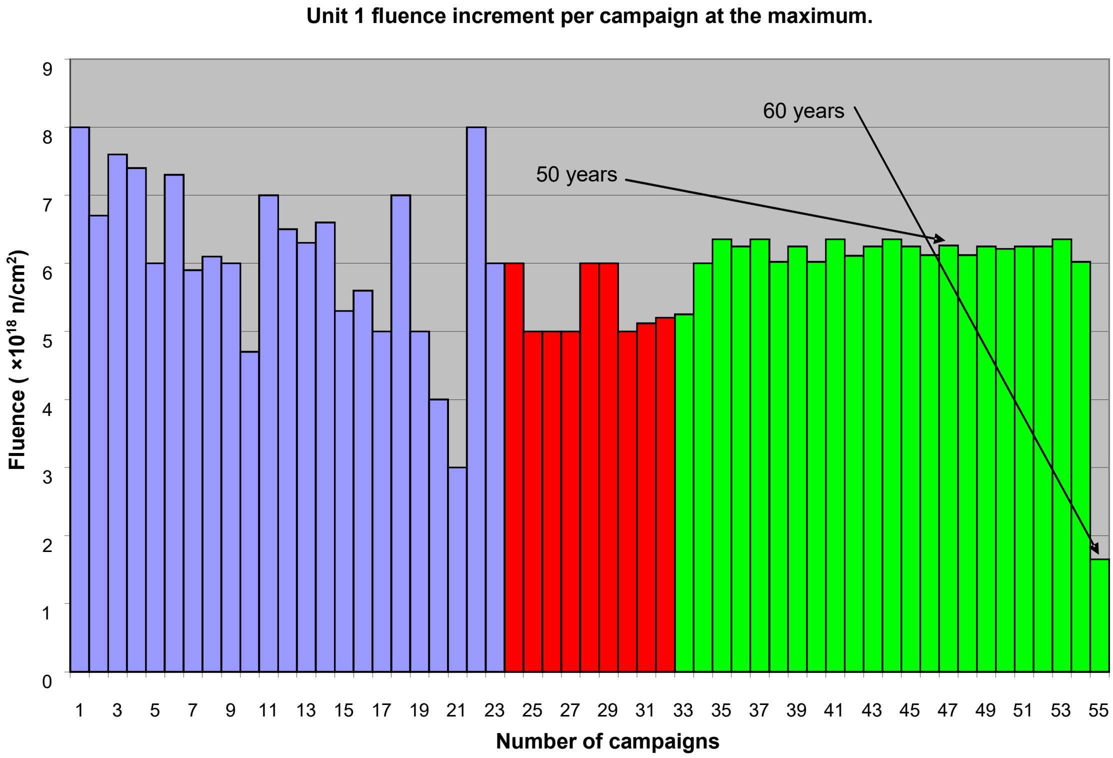

Figure 05.2.2-1: Change of fluence values on the RPV of Unit 1 of Paks NPP [48]

05.2.3. Conclusions of the licensee related to the ageing management of RPVs

The ageing management practice of RPVs is in line with the national regulations and the relevant international requirements, recommendations. This is supported by the full scope of activities in the framework of or closely related to the above mentioned RPV SAMP-001 according to the review aspects of the WENRA AM National Report. In addition, SALTO, IAEA, TC and US independent peer reviews conducted during the preparation for the service life extension licensing also draw this conclusion.

Modifications of the RPV ageing management programmes contributed effectively to the prevention, management of the concerned degradation mechanism.

The management of the failure due to the fatigue of the welds of the CRD nozzle inner tubes is the only example, where the ageing management despite several programme modifications, cannot clearly exclude the possibility of further damage to the so far unrepaired welds. At the same time through the applied detection practice, the occurrence of the latter case is detected in a timely manner to maintain the component's safety function, and following the restoration using improved repair technologies, it is expected that there will be no further weld failures on the nozzles concerned.

Budapest Research Reactor

The vessel of the research reactor is not a pressure retaining equipment, it is atmospheric, thus it does not belong strictly to the scope of the review, however, for the sake of a more complete picture the information was included here.

Ageing management of the reactor vessel

	Inspection, testing, monitoring	Evaluation
1	Accelerated ageing material testing 1993-1996	Reactor Safety Committee
2	Primary water parameters, dripping detection	Reactor operator
3	Visual inspection after core reconfiguration	Reactor operator
4	Water sampling analysis quarterly	AMP head, reactor operator
5	Structural inspection and pressure test every 4 years	AMP head, reactor manager
6	Visual inspection + video recording annually	AMP head

Ageing assessment of the reactor vessel

Before reconstruction of the Budapest Research Reactor the Soviet reactor designers were consulted concerning the material of the new vessel. The Soviet designers did not recommend to construct the vessel from SZAV-1 type AlMgSi alloy due to welding problems and embrittlement of the welds despite the fact that this alloy performed well as the material of the first vessel for 30 years of operation.

The first proposal was to use Al99.5 quality unalloyed aluminium. However, it would have had more disadvantages: relatively soft material, its quality and corrosion-resistance properties have large deviations, since it the impurities vary, and strong activation is anticipated because of the high impurity content, which makes maintenance difficult and increase the amount of radioactive waste.

At this point the BRR turned to the Metallurgy Research Institute of Hungary. After several consultations the institute proposed to use the eutectic composition variant of the AlMg2.5 alloy manufactured on 99.95 aluminium basis. This material was named R-AlMg2.5. The AlMg5 type electrode was selected for welding. After selecting the material two experimental batches were manufactured, on which a set of qualification tests were carried out by the institute, the BRR and the Kurchatov Institute. The Metallurgy Research Institute performed in 1983 strength tests on these batches, elaborated the welding technology and confirmed the corrosion properties.

The BRR prepared large number of specimens from the material of the previous vessel and the new vessel and from the welds. A part of these were irradiated in the research reactor with a fluence of $3*10^{19}$ n/cm^2 and then tested. In parallel the Kurchatov Institute also examined the samples with an irradiation fluence of $2*10^{20}$ n/cm^2. Both examinations demonstrated the excellent ductility and radiation resistance of the R-AlMg2.5 alloy and the welds, significant embrittlement was not experienced. Si was deposited on the grain surfaces and the impact energy decreased to the half in the case of the SZAV-1 weld, but it still remained sufficiently ductile to safely operate.

A formerly operated and dismantled reactor vessel was also examined. No damage (flaw, corrosion) was detected on the vessel, the tests of materials of the internals also demonstrated that neither corrosion nor cracking occurred, and that mechanical properties were appropriate. The results were published at the IAEA and conferences.

<u>Current state of the reactor vessel and the structural components:</u>

The following effects were taken into account in the ageing management of the reactor vessel material:
a) corrosion of the structural material;
b) thermal ageing of the structural material;
c) low and high cycle fatigue of the structural material;
d) radiation damage of the structural material.

Corrosion:

The R-AlMg2.5 alloy is one of the best corrosion resistant aluminium alloys. There are several decades of experience with similar alloys in Hungary (e.g. a similar, but not so clean aluminium alloy named Nautal was produced by KÖFÉM that is still used in boat industry for water buses and yachts, and for decades there is no significant corrosion in contaminated surface waters. There are two locations where corrosion is possible on the vessel of the research reactor: on the bottom if an iron object is dropped at it remains there undisturbed (this is not possible because of the high flow rate in the VVRSz-M type reactor) and near the water surface where the environment of the material is changing between air and water. The coolant of the Budapest Research Reactor is pure, transparent water, so in the case of each core reconfiguration a thorough visual inspection of corrosion is carried out under the head and near the water surface. During the examination till no sign of corrosion was detected (surface colouring, hole, corrosion product deposit etc.).

Thermal ageing of the structural material:

Thermal ageing of the structural material is the last phase of segregation hardening. This is very typical for AlMgSi alloys, but not typical for eutectic composition AlMg2.5 alloys and welds. Thermal ageing at room temperature also completes in 1-2 years, and sooner at the operational temperature of the reactor. The surveillance specimen set of the reactor (see in detail in Radiation damage of the structural material) was in the reactor for 2.5 years after restart. Thermal ageing could then be monitored by the specimen set together with radiation damage.

Low and high cycle fatigue of the structural material:

Load of the vessel is almost static, the design load is the weight of the medium contained in it. Complete emptying and refilling took place only three times since the commissioning. Therefore no low cycle fatigue should be taken into account. The vessel does not vibrate during operation, consequently high cycle fatigue can also be neglected. Vibration of the pipelines were tested during commissioning by acceleration measurements, and based on the results the load of the pipelines and the nozzles is far below what could cause fatigue of the components.

Radiation damage of the structural material:

A surveillance program was elaborated by the BRR to monitor the design data for radiation damage and thermal ageing and for planning of reconstruction. The specimen set according to the programme was manufactured from the same material with appropriate markings. The following specimens were placed in the reactor before start up:

- 16 pieces of 10*10*55 mm Charpy V-notch specimens made of the material of the vessel sheet. Marking: E.
- 16 pieces of 10*10*55 mm Charpy V-notch specimens made of the material of the test sheet manufactured for the longitudinal weld of the belt at the core level. Marking: V.

- 16 pieces of 10*10*55 mm Charpy V-notch specimens made of the material of the horizontal channel 4. Marking: Z.
- 16 pieces of 10*10*55 mm Charpy V-notch specimens made of the material of the core grid. Marking: X.

8-8 of the above specimen sets were encapsulated in one-one standard irradiation capsules. Water cross-flow was ensured through holes, thus the surface of the specimens was wetted. The specimen spent 36 months in the reactor, and no corrosion signs were found on the surfaces of them after taking out. Total irradiation time was 6216 hours, the calculated fluence was $3.38*10^{20}$ n/cm^2 E>1 MeV and $4.95*10^{20}$ n/cm^2 E>0.5 MeV volt. Irradiation temperature was equal to the service temperature of the reactor.

The same number of unirradiated specimens were provided and tested in parallel, so a direct comparison could take place.

- **Charpy tests:**

The examinations were implemented in the material testing laboratory of the BRR on the specimens taken out. The tests were 300 J standard, accredited impacts tests at 20 and 80 °C temperature (which is in compliance with the two boundary values of the operational temperature).

The test results are summarized below:

Results of Charpy tests

Material	Marking	Fluence [10^{20} n/cm^2 E>0.5 MeV]	Test temperature [°C]	Charpy energy* [J]	Dynamic yield point **
Vessel wall	E	0	20	101	78
			80	91.75	71
		4.95	20	149.2	101
			80	149.0	101.6
Core grid	X	0	20	107.3	69.5
			80	104	71.5
		4.95	20	148.5	86
			80	155.5	92
Horizontal channel	Z	0	20	109.0	73.5
			80	102.0	66
		4.95	20	142.5	95.4
			80	139.0	107
Weld	V	0	20	60.3	74
			80	56.3	71.5
		4.95	20	62.5	102.5
			80	51.3	103

*Average measured on 4 specimens **Calculated from Charpy curves

Conclusions on the vessel ageing

Concerning the materials applied in soft conditions (core grid, horizontal channel) ductility increased significantly, while in the case of the rolled wall material and the welds the changes remained within the Charpy test deviation, that is no damage could be identified. All measured values exceeded the 27 J average and the lowest impact energy measured for a specimen was double of the required minimum of 20 J.

Evaluation of the instrumented impact test showed concerning the irradiated specimens an increase of the dynamic yield-point and a decreased deformation till cracking. Cracking took place in all the cases after significant plastic deformation, a considerable number of the specimens only bent and not cracked. No instable crack propagation could be identified. The specimens were located in direct vicinity of the core, that is the calculated and measured irradiation levels significantly exceeded the fluence calculated for the vessel wall and the welds (including the welds of the core grid) for the 30 years of design lifetime ($3x10^{20}$ n/cm^2 E>0,5 MeV). It should be noted that the design documentation calculated with 20 MW power, but the reactor was operated since the reconstruction at only 10 MW, which means 50% less damage. Based on the results, in accordance with the design objectives it can be concluded that the vessel can operated for at least 30 years from strength and integrity point of view.
Processing the data of near ten years of operation it can be concluded that no corrosion or corrosion like phenomenon could be detected on the vessel components. Radiation damage or thermal ageing was not detected on the surveillance specimens of the reactor. Low and high cycle fatigue should not be considered due to the low cycle numbers and low level of mechanical vibration loads. According to the tests carried out with fluence values corresponding to 30 years of operation the vessel can operate at least for 30 years from strength and integrity point of view.

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

The HAEA reviewed the information submitted by Paks NPP and compared to that information and data obtained and provided during its regulatory activities, first of all during the licensing and inspection performed for service life extension. The conclusion is that the description is in agreement with the knowledge of the HAEA.

It can also be declared that the ageing management practice applied for the reactor pressure vessels of the facility is in compliance with the Hungarian regulations and so the international requirements. Paks NPP manages all required and known ageing mechanisms of the reactor pressure vessels, monitors and is prepared for the management of the potentially occurring degradation mechanism. The modifications of the reactor pressure vessel ageing management programme efficiently contributed to the prevention and management of the concerned degradation mechanisms in line with the requirements. The activities performed within or associated with the ageing management of the reactor pressure vessels ensure that the cycle numbers determined by the strength calculations will not be exceeded.

Paks NPP developed a supplementary surveillance programme to satisfy the regulatory requirements for the extended service life period of the units and to more closely follow the condition of the reactor pressure vessel. This makes possible to even better monitor the neutron bombardment and to refine the validation of the calculations. Actual evolution of the fluence, and results of the actions will be reassessed during 2019-2020 after the testing of the specimens.

With respect to the recent international experience it should be noted that internal and external visual and ultrasonic and internal surface eddy current testing of the reactor pressure vessels of Paks NPP are parts of the in-service inspection programme. A feature of these examinations is that the scope covers the welded joints, the weld environments and the belt line material in front of the core. Non-destructive examinations are qualified according to the ENIQ methodology.

Ageing management practice of the Budapest Research Reactor complies with the expectations and national requirements. Tracking and assessment of the parameters are appropriate and evaluation on life time was established and adequate. The authority accepts the conclusion of the Licensee that the service life of the reactor vessel is at least 30 years.

06. Calandria/pressure tubes (CANDU)

Not applicable, in Hungary there is no CANDU type reactor.

07. Concrete containment structures

Paks Nuclear Power Plant

07.1 Description of ageing management programmes for concrete structures

The containments at Paks NPP are reinforced concrete structures. The scope of ageing management applied in the field of buildings, structures, the commodity groups based on material and structure and the structural commodity groups were already discussed in Section 02.3.1.3.

Building- and structure-programmes have been developed for the ageing management of the concrete containment structures. Both types of programmes use the same structural commodity groups and main groups. The structure-programmes contain the degradation mechanisms of the structural commodity group, the expected degradation effects and their management methods. The building-programmes complement the structure-programmes for a given building in that they define the criteria, framework, logistical aspects etc. of the implementation of the structure-programmes. In line with this, the building-programmes contain building specific requirements related to the critical areas of the structural commodity groups; and require supplementary ageing management activities not or not fully managed in the structural commodity groups.

07.1.1 Scope of ageing management for concrete structures

07.1.1.1. Method and criteria for determining the scope of AM of concrete containment structures

The layout of the concrete containment is shown in Annex 07.1.1.1-1 of this report. The ageing management of the concrete containment structures is realized in the framework of the BAMP-B-002 building-programme, as well as the associated structure-programme in accordance with Table 07.1.1.1-1

Table 07.1.1.1-1: List of structure-programmes used for concrete containment structures

ID of structure AMP	Title of structure-AMP
BAMP-A-002	Foundations AMP
BAMP-A-006	Pipe and cable support structures AMP
BAMP-A-016	Doors and hatches AMP
BAMP-A-018	Stainless steel plates AMP

ID of structure AMP	Title of structure-AMP
BAMP-A-013	Leak-tightness of the hermetic compartment AMP
BAMP-A-001	Steel structures AMP
BAMP-A-004	Coatings AMP
BAMP-A-005	Surveillance of boric acid corrosion of reinforced structures AMP
BAMP-A-008	Monitoring of building settlement AMP
BAMP-A-011	Machine foundation degradation caused by fatigue AMP
BAMP-A-012	Hermetic lining AMP
BAMP-A-015	Concrete structures exposed to elevated temperature AMP
BAMP-A-019	Seals AMP
BAMP-A-020	Fire barrier structures AMP
BAMP-A-022	Electrical distributors, control and relay panels, local actuation cabinet AMP
BAMP-A-021	Reinforced concrete superstructures AMP

The above-mentioned AM programmes are related to the following structures:

Reinforced concrete structures (regular concretes and biological protection concretes), including reinforcing steel and embedded steel insertions as well)

- Reinforced concrete basemat and walls in the underground/soil environment;
- Reinforced concrete superstructures, walls, slabs exposed to the air outdoor;
- Machine bases in substructure environment and inside the building;
- Concrete superstructures, reinforced concrete walls and slabs in air indoor.

Steel structures (carbon steel and low-alloyed steel, in the environment outside the building)

- Steel frame bridges between the localisation towers of twin units. (These supports were installed for improving seismic resistance)

Steel structures (carbon steel and low-alloyed steel, in the environment inside the building)

- Remaining “steel cell formwork of” walls and floor slabs, steel support structures and liners including line anchorages;
- Hermetic liner;
- Steel supports and their bolted connections;
- Doors (biological protection doors and fire barrier doors), steel elements and weld seams of the penetrations;
- Steel support structures of mechanical and electrical equipment.

Steel structures (stainless steel, in environment inside the building)

- Compartment liners;
- Drainage lines.

Coatings

- Decontaminable coatings;
- Anti-corrosion coatings ;
- Fire protection coatings.

Seals

- Sealing elements of doors and hatches;
- Strip seals for expansion joints;
- Seals of liners, connections

The above mentioned ageing management programmes of Paks NPP fully cover the scope of the containment structure according to Guideline 4.12 [15]

07.1.1.2. Determination of the degradation mechanisms

Degradation mechanisms were determined in line with Guideline 4.12 [15], company and international operational experience, and relevant international literature. Detailed overview of these are included in the Hungarian documents [49] and [50] with appropriate references. Degradation mechanisms can be accounted for in the programmes that can be identified via periodic reviews, adapting the information of the latest IAEA, ACI, US NRC, EPRI etc. documents and the experiences of the relevant international AM practice.

07.1.2 Ageing assessment of concrete structures

The following subsections have been compiled considering the specifics of the reactor buildings of the VVER-440 NPPs, based on the examples of ageing management practice of the recommended sample components of reinforced containment structures, such as

- The concrete used in the concrete containment;
- The steel reinforcement;
- The prestressing system[3];
- The liner;
- Connections between liner and the concrete structure such steel elements of the liner anchors;
- Waterstops, seals and gaskets and protective coating.

07.1.2.1. Ageing mechanisms and acceptance criteria

The degradation mechanisms of the containment (including reinforcements), hermetic lining, steel parts, waterstops, seals, gaskets, protective coatings are described in the building programme BAMP-B-002 for the main reactor building, the acceptance criteria of the structural AMPs assigned to it are presented in Table 07.1.1.1-1as follows.

Concrete (including steel reinforcements as well)

The relevant ageing effects are **material loss, cracking** and **change in the material properties**, that are associated with the degradation mechanisms: **freezing-thawing, corrosion of steel reinforcements, differential settlement, thermal exposure, fatigue, Ca(OH)2 leaching, chemical attack, irradiation.**

The acceptance criteria are as follows:

- The strength of the concrete must correspond to the design values;

[3] No pre-stressing systems were used at the Paks NPP.

- The crack widths during exposure to various environmental effects should not exceed the values contained in documents MSZ EN 1999-1-1:2007/A1:2010 [51] and ACI 349.3R-02 [52];
- Corrosion of the steel reinforcements is not permitted;
- Classifications based on the surface defects and their sizes into the various degradation grades, must be done based on document ACI 349.3R-02 [52];
- Depending on the results of the visual inspection, field tests or, core sampling and laboratory testing maight be required. Special acceptance criteria exist for the results of each test.

Heavy concrete structures with shielding function exposed to the attacks of borated water have additional criteria, for example:

- There should be no spalling, no cavities filled with stagnant liquid on the concrete surface;
- The coverage of the steel reinforcement should be at least 30 mm everywhere;
- Surface material loss should not exceed 3% for the entire surface;
- The boron content should not exceed 1% of the concrete content.

Hermetic lining

Ageing effects are **material loss, cracking, mechanical deformation** and **change in the material properties**, that are associated with the following degradation mechanisms: **general corrosion, crevice corrosion, boric acid corrosion, irradiation.**

The acceptance criteria are as follows:

- Acceptability of the condition of the coating;
- Corrosion related material loss up to a maximum of 10%;
- Cracks should not occur.

Steel product

Ageing effects are **material loss, cracking,** and **change in the material properties**, that are associated with the following degradation mechanisms: general corrosion, crevice corrosion, stress corrosion, wear, irradiation.

The acceptance criteria are as follows:

- Acceptability of the condition of the coating;
- Material loss should not exceed 4% of the required structural thickness;
- Cracks should not occur.

In case of material loss exceeding 4% of the required structural thickness experts must be involved for evaluation of condition.

Water stops, seals, gaskets

Ageing effects are **material loss, cracking,** and **change in the material properties** that are associated with the following degradation mechanisms: **mechanical effect, chemical attack, microbiological attack, thermal exposure, irradiation.**

The acceptance criteria are as follows:

- No mechanical damage, or damage caused by burning, no cracks;
- The original base colour should not change ;
- Free from fungus and pollutants;

- Shore hardness may deviate from the baseline value by ±10%

Coatings

Ageing effects are **material loss, cracking,** and **change in the material properties**, that are associated with the following degradation mechanisms: **mechanical effect, chemical attack, microbiological attack, thermal exposure, irradiation.**

The acceptance criteria are as follows:

- Fading, dulling not permitted;
- No mechanical damage, or damage caused by burning;
- Cracking, blistering, delamination is not permitted;
- Fungus or pollutants are not permitted;
- In instrumental testing the adhesive strength should be at least 0.7 MPa for concrete surface, at least 2.5 MPa for steel surface.

07.1.2.2. Standards, guides and manufacturing documents

The following documents were taken into consideration in the ageing management of concrete containment structures:

- HAEA Guideline 4.12 [15];
- IAEA, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Concrete Containment Buildings. IAEA-TECDOC-1025, 1998. [7];
- IAEA, Maintenance, Surveillance, and In-Service Inspection in Nuclear Power Plants, IAEA No. NS-G-2.6, 2002. [53];
- IAEA, Safety Aspects of Long Term Operation of Water Moderated Reactors. Final Report, 2007. [54];
- IAEA: Guidebook on Non-Destructive Testing of Concrete Structures, IAEA-TCS-17, 2002. [55];
- ACI 349.3R-02, Evaluation of Existing Nuclear Safety Related Concrete Structures. [52];
- ACI 201.1R-92. Guide for Making a Condition Survey of Concrete in Service [56];
- SEI/ASCE 11-99. Guideline for Structural Condition Assessment of Existing Buildings. [57];
- US NRC, Generic Aging Lessons Learned (GALL Report). NUREG 1801, 2010. [3];
- ASME XI. Subsections IWA, IWB, IWE, IWF, IWL. [11];
- ASTM D5163-05a. Standard Guide for Establishing Procedures to Monitor the Performance of Coating Service Level I Coating Systems in an Operating Nuclear Power Plant, 2005. [58];
- Aging Effects for Structures and Structural Components (Structural Tools) Revision 1. EPRI, Palo Alto, CA: 2003. 1002950. [59];
- The Effect of Elevated Temperature on Concrete Materials and Structures. A Literature Review. Rep. NUREG/CR-6900, 2006. [60].

07.1.2.3. R&D programmes used in the ageing management of concrete containment structures

As an example of the R&D programmes used in the ageing management of concrete containment structures, the programme aimed at the condition assessment of regular and heavy weight concrete including core sampling is presented.

Assessment of the long-term behaviour of structures and structural elements made from concrete and regular concrete, destructive and non-destructive tests are conducted in order to better understand the actual strength as well as the ageing mechanisms of the steel reinforcements and liners.

In the framework of a programme consisting of evaluation of two typical heavy concrete types (concrete containing hematite or hematite and steel pellets) and normal gravel concrete with specific density built into the hermetic compartment in rooms A201/1-1 and A 201/3-3 of Units 1 and 3 located at +6.00 m, samples have been taken from the steel plate covering the concrete and on-site inspection of the exposed concrete surfaces have been carried out. Core samples were also taken and their laboratory testing was carried out. The sampling activities did not involve the opening of the hermetic lining. The on-site inspection of the concrete and steel reinforcements consisted of the following:

- Visual inspection;
- Schmidt-hammer testing;
- Adhesion tests;
- Determination of the pH values;
- Determination of the hardness of the steel reinforcements;
- Inspection of the diameter of the steel reinforcements;
- Hardness measurement.

The laboratory tests of the core samples determined the following characteristics:

- Compressive strength;
- Density, bulk density, porosity;
- Foreign material content in the concrete;
- Condition of the steel cladding plates.

The tests provided important results about the actual strength characteristics of the concretes, as well as the ageing mechanism of the concretes, steel reinforcements and steel cladding.

The results of these tests were also used in the building ageing management programmes, structural ageing management programmes as well as in the TLAAs titled ‘Static and strength lifetime analysis of important buildings’ and ‘Lifetime analysis of material property change of heavy weight concrete structures’.

07.1.2.4. Consideration of operational experience

The AM programmes require consideration of internal operational experience, and relevant foreign and domestic non-nuclear power plant experiences. To facilitate the overview of international practice and experiences documents have been developed (exists in Hungarian) that present a comprehensive view of the entire field (see in Section 07.1.1.2). Furthermore, documents focusing on certain areas have also been developed (e.g. concerning VVER containments) and some other documents are also being developed e.g. on the effect of borated water leaks on reinforced concrete structures etc. Utilizing own operational experiences, it is important to select and update the critical areas to which special attention must be paid during the inspections, including the determination of the progress of degradation mechanisms.

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

This Section deals with the following two areas: Activities, methods, frequency of inspections (Subsection 07.1.3.1.) and Consideration of inspection history (Section 07.1.3.2.). The acceptance criteria belonging to the different ageing mechanisms and ageing effects are presented in Subsection 07.1.2.1.

07.1.3.1. Activities, methods, frequency of inspections

For the components in Section 07.1.2 the following inspections and inspection frequencies are applied:

Concrete (including steel reinforcements)

- Measurement of the building settlement, annually.
- Analysis of chemical content of the groundwater, annually.
- Visual inspections, at least once every 5 years. The inspections also include mapping the leakage traces of borated water. If the results of the visual inspections require, instrument tests, core sampling and their analysis are carried out.
- Analysis of cracks, crack monitoring, crack mapping, every 2 years.
- Cyclic testing of the regular and heavy concrete structures exposed to the effects of borated water leakage at least once every 5 years, at the 3-3 inspection hatches of each Unit. These tests consist of visual inspection, field instrumental tests, borehole sampling and laboratory testing.

Hermetic lining

- Integrated leakage test, every 15 months.
- Local leakage test, every 15 months.
- Visual inspection, with ultrasonic thickness measurement of the liner to detect material loss, every 5 years.

Steel parts

- Visual inspection, every 5 years.

Water locks, seals, gaskets, coatings

- Visual inspections, every 5 years. If the results of visual inspections require, instrument tests (measurement of adhesive strength, hardness etc.) are carried out.

A significant portion of the above mentioned activities are carried out by ‘third party certified organizations’, which are expert organizations of the field. Detailed Ageing Management Report documents are prepared about the tests and evaluation of the results.

07.1.3.2. Consideration of inspection history

Typical examples of trending and gradual degradation are as follows:

- cracking monitoring with updating the crack maps in case of reinforced concrete structures;
- building settlement monitoring and evaluation;
- regular inspection and evaluation of the effects of borated water leakage appearing in the concrete using inspection hatches etc.

Important are the following activities:

- identification of additional and unexpected degradations;
- utilization of the latest international and own experience;
- expert analyses based on on-site inspections;
- consideration of the experiences in non-civil engineering fields, etc.

07.1.4 Preventive and remedial actions for concrete structures

07.1.4.1. Preventive actions

The preventive actions are contained in the relevant building-programmes and structure-programmes.

Some typical examples related to concrete containment structures:

- Repair works of minor defects detected during the integral leakage test of the hermetic liner, repairs at the air-lock doors identified via depression measurements.
- Regular inspection of the chemical composition of the groundwater to assess its agressivity regarding the reinforced concrete substructures (based on current results, the groundwater is not aggressive at the Paks NPP, the pH value, the concentration of chloride ions and sulphate ions are far from the aggressive limit).
- Monitoring, minimization and elimination of borated water leaks seeping into the reinforced concrete structures, which aids the management and prevention of damage of the regular and heavy concrete structures (including the steel reinforcements) due to the borated water.
- Maintaining the coating of steel structures in good conditions to prevent corrosion.
- In the case of reinforced concrete and steel structures, as well as the coatings, the early repair is an important preventive action.

Timely reparation of minor defects of the reinforced concrete and steel structures as well as the coatings is an important preventive action as it can prevent the formation of larger defects.

The implementation of the appropriate preventive actions must be initiated by the technical Department/Section managing the structure. The actions to be taken must be based on the requirements related to the maintenance practices of the plant and the consideration, management of individual expert opinion. The inter-organizational responsibilities are described in Section 02.3.1).

07.1.4.2. Remedial actions

The various corrective actions are contained in the relevant building-programmes and structure-programmes. In more complex cases, for the preparation for actions, individual expert opinion, evaluations, detailed design documents are developed.

Typical examples related to concrete containment structures:

- Patch-like repair of decontaminable coatings or implementation of new coatings. The implementation is performed using alrady existing standardized procedures.
- Patch-like repair or replacement of carbon steel cladding. The implementation is performed using alrady existing standardized procedures.
- Repair of the hermetic liner to eliminate corrosion damages. The implementation is performed using alrady existing standardized procedures. (A significant degree of local corrosion damage has occurred at one location in Unit 3. Here, the liner is on the upper surface of the reinforced concrete slab and is covered by a concrete layer. A pipe transporting borated water from the spent fuel pool is located inside this concrete layer. Due to the leakage of this stainless-steel pipe borated water of high oxygen content flowed directly onto the hermetic liner and this resulted in a wall-through hole of approximately 52 x 24 mm. The repairs were done by welding a new plate on.)
- Repair of concrete surface damage, steel cladding and anchors. The most significant damages so far have occurred on the bottom surface of the heavy concrete floor slabs of rooms A301/2-2 and A301/3-3. These slabs were made using carbon steel girders with a height of 800 mm and the girders were welded together on site. The steel rings ensuring the placement of the main circulating pump and main gate valves, which were welded to the steel girders using steel structures and were fixed into the concrete of 3650 kg/m3 specific density filling the space between the girders and the rings and reinforced at the top and bottom. Based on the design documents and taking into account the results of the condition analysis, heavy concrete was injected into the damaged areas, and the carbon steel protective cladding and its anchoring were also replaced. Considering the results of the ongoing evaluations and analyses, the relevant ageing management programmes will be updated.

Main characteristics of the requirements concerning corrective actions are the following:

- Decisions made about the technical and/or administrative actions must be based on the expert analysis of the conducted inspections.
- Some structures require repairs or restoration in the following cases:
  - If further damage is expected based on the inspections that may endanger the structure’s design function prior to the next inspection.
  - Trends indicate deviations from the acceptance criteria.
  - The structure has already been damaged to such an extent that endangers the fulfilment of its design function.
- The implementation of the corrective actions must be initiated by the Department/Section managing the building or, structure.
- The actions to be taken must be based on the requirements related to the maintenance practices of the plant and the consideration, management of individual expert opinion. The inter-organizational responsibilities are described in Section 02.3.1).

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

07.2.1. Assessment of degradation mechanisms to be managed

The assessment of the development and management of the degradation mechanisms of the concrete containment structures is summarized in connection with indication of the structure-AMPs of Table 07.1.1.1-1.

1. Foundations

No signs of degradation mechanisms were detected at the building foundations. (The groundwater is not aggressive).

2. Pipe and cable support structures

Degradation mechanisms are not typical, prevention and repair of small defects is done through regular maintenance.

3. Doors and hatches (protection doors)

In the case of biological protection doors degradation mechanisms are not typical, their prevention is done through standard inspections and maintenance (e.g. in some places the threaded locking mechanism has been replaced). Damage to the fire protection doors, mainly resulting from use, is done on a regular basis.

4. Stainless steel plate

There is only a negligible amount of local damages on the stainless-steel plate coverings of the rooms.

5. Leak-tightness of the hermetic compartment

According to the integrated leakage tests, the leak-tightness of the containment is adequate and shows only minor changes. Identification of smaller local leakage areas (e.g. at the seals) and the elimination of the identified leaks are done on a regular basis.

6. Steel structures (carbon steel and low-alloyed steel)

Degradation mechanisms are not typical, prevention is done through visual inspections and maintenance (repairs of the coating or implementation of new coating as necessary).

7. Coatings

The decontamination property of the decontaminable coatings has decreased slightly, but its reparation is done on an annual basis as part of regular maintenance.

8. Boric acid corrosion of reinforced concrete structures

The rate of borated water seeping into the reinforced and the heavy concrete for shielding has so far only been reduced, but not completely eliminated. The intensity of the leaks is the highest during refuelling; the leaks are probably originating from the pools or shafts and their related components. During ageing management, the traces and paths of the leaks are monitored. 3-3 inspection hatches in every unit were created at the regular concrete and heavy weight concrete structures in the areas most susceptible to leakage, through which inspections and assessments are carried out by every 4 campaigns to follow the effects of borated water leakage on concrete, steel reinforcements and steel cladding.

In addition, in several places further core sampling and inspections of regular and heavy concrete structures have taken place. Operational experiences show that the degradation effects are faster in the case of heavy concrete than in the case of regular concrete. The most significant damages so far have occurred on the bottom surface of the heavy concrete floor slabs of rooms A301/2-2 and A301/3-3. (The main information related to the design of these slabs is contained in Section 07.1.4.2). Based on the design documents taking into consideration the results of the detailed condition analysis, heavy concrete was injected into the damaged areas, and the carbon steel protective cladding and its anchoring were also replaced. Taking into consideration the results of the currently ongoing evaluations and analyses, the relevant ageing management programmes are expected to be updated.

9. Building settlement

Due to the nature of the soil and the seasonal change of the groundwater level, the main reactor builings are affected by a continuous settlement process. The acceptance criteria of settlement for the measuring points have been determined by analysis of the integrity and function of the reactor building and prognosis of the settlement. The measured settlement does not endanger the function of the building. The measurements show that the consolidation process of the soil has slowed down considerably, now mostly only the impact of seasonal fluctuations in ground water level affecting the vertical movements.

10. Machine bases

Degradation mechanism is not typical for the plant, prevention of the processes and repairs of minor defects are done through regular maintenance.

11. Hermetic liner

On the accessible side of the hermetic liner at the joints of the various structures the coating is cracked or damaged in some places. Here general surface corrosion occurs. If borated water leakage is also present in these areas, boric acid corrosion occurs. Repair of these defects is done regularly. According to the ultrasonic thickness measurements taken on the accessible side of the hermetic liner, the decrease in the thickness of the cladding in most areas is less than 10% which is the acceptance criteria. Thickness reduction more than 10% occurs locally in relatively few areas, the highest value is less than 15%. Control calculations done for the areas of deviations show that the hermetic liner fulfils its function.

12. Concrete exposed to elevated temperature

Degradation mechanism is not typical for the plant, the higher temperature penetrations are cooled.

13. Seals

The lifetime of seals (so long that they can fully carry out their functions) is generally in line with the general requirements. Worn-out, defective seals are replaced (e.g. in the case of biological protection doors, maintenance openings etc.)

14. Fire barrier structures

Degradation resulting typically from wear are moderate, minor repairs are done during maintenance.

15. Electrical distributors, control and relay panels, local actuation boards

Degradation mechanisms are not typical for the plant, prevention of the processes and repairs of minor defects are done through regular maintenance

16. Reinforced concrete superstructures (regular concrete and heavy concrete)

The most significant degradation mechanisms occurred due to borated water leakage (See no. 8 of this list above). In addition, there are occasional minor surface defects, cracks (practically all cracks are dormant cracks). These are monitored regularly, repaired as necessary, primarily to protect the steel reinforcements against corrosion.

07.2.2. Modifications in the programmes and their justification

Section 02.4 contains a detailed description of the review and modification of the overall AMP. These generally apply to concrete containment structure programmes as well. In the following, therefore only one typical example of an already implemented AMP modification will be presented, and another example for a planned modification.

Major changes to the AMP were carried out in the monitoring of building settlement AMP (BAMP-A-008) related to the operational building settlement examination and assessment. The reason for the modification was to introduce acceptance criteria defined by strength analysis using a verified soil-building model. The modified criteria are better fit to the actual settlement processes.

Operational experiences show that the degradation effects are faster in the case of heavy concrete than in the case of regular concrete, especially at the bottom surfaces, where the seeping water can accumulate. Taking into consideration the results of the currently ongoing evaluations and inspections, the relevant ageing management programmes are expected to be updated.

07.2.3. Conclusions of the licensee related to the ageing management of concrete containment structures

The ageing management practice of concrete containment structures is in line with the national regulations and the relevant international requirements, recommendations, as supported by the findings of the SALTO, IAEA, TC and US independent peer reviews conducted during the preparation of the service life extension licensing. During the reviews only minor modifications, amendments were recommended, which were implemented. For example, the chapters titled ‘preventive and corrective actions of ageing mechanisms’ and the sections dealing with the management of cracks of the reinforced concrete structures have been expanded, and the newer sections of the amended ACI 201.1R document [56], have been taken into consideration for the visual inspections of reinforced concrete structures.

Experience has shown that the detailed Ageing Management Reports, which contain in addition to the inspection results expert assessments and suggestions too, contribute effectively to the prevention and management of the degradation mechanisms, as well as to the further improvement of AMP.

Examination, analysis and management of the consequences of the borated water leakages on reinforced and heavy concrete structures remain important tasks in the future.

Operating experience points out that the degradation mechanisms are faster in the case of heavy concrete structure compared to the normal concrete structure, especially at the lower parts of horizontal surfaces the most subjected to the effects of leakages. In the rooms A301/2-2 and A301/3-3 there were such damages at certain locations, which have been repaired by injecting heavy concrete and replacement of the carbon steel cladding and anchoring after removal of the damaged parts.

Based on the analyses and examinations in progress, improvements of the related ageing management programmes are expected. In addition, the activities aimed at determination and elimination of sources of borated water leakages need to be continued and further improved in terms of effectiveness.

Budapest Research Reactor

Not applicable, it does not have a containment.

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

Evaluation of ageing management submitted to the HAEA by Paks NPP in the frame of regular reports contains the ageing management report on the reinforced containment structures. The HAEA regularly comments these reports, and according to the annual inspection plan conducts inspections, if it is necessary or justified. The compliance, scope and mode of application of the ageing management programme are usually inspected during these inspections.

In the service life extension process the HAEA reviewed the efficiency of the ageing management programmes. During 2017, in the frame of assessment of the license application for Unit 4 service life extension the HAEA put emphasis on the conditions of reinforced containment structures, to which an external expert was also involved. The evaluation covered the assessment of the ageing management activity and the organizational issues.

In general, the conclusion was that the required scope is covered by the ageing management programme of Paks NPP. The ageing management practice complies with the national regulations and recognized international requirements.

The HAEA revealed smaller deviations in the ageing management programme that were corrected by the Licensee. A proposal was made at the same time to issue a new regulatory guideline to explain the requirements on the content and form of building and structure related ageing management reports.

08. Pre-stressed concrete pressure vessels (AGR)

There are no such components in the Paks NPP and in the Budapest Research Reactor, thus this chapter is not applicable.

09. Overall assessment and general conclusions

Regulation of ageing management in Hungary is organic part of the nuclear safety requirements from 2005. Accordingly, in the case of both concerned nuclear facility types there are detailed requirements on design and operation in the Nuclear Safety Code regarding this area. The requirements are further explained in the case of the NPP in the regulatory guidelines focusing on the design and operation issues of ageing management. Ageing management therefore is embedded in the regulatory oversight process as an individual technical area.

The permanent nuclear safety oversight implemented by the HAEA uses all the regulatory instruments determined by the Act on Atomic Energy. Content requirements for the specific license types determine the ageing management related expectations according to the given life cycle phase of the facility or to the given activity/modification. The HAEA reviews the compliance in the frame of the given licensing process. As a designated technical area for regulatory inspections, the ageing management is a grave aspect of the ten yearly due periodic safety reviews. In the regulatory assessment process, based on the regular reports, event investigations, licensing and inspection acts, the HAEA also scrutinizes the ageing management activity of the Licensees.

It is to be emphasized that both in the requirements and in the regulatory oversight process, the consideration of a graded approach is a very important aspect. Accordingly, as the various facilities as the safety classes within the various facilities are significant viewpoint in relation to differentiating the requirements and the applied depth and instruments of oversight.

Using the results of regulatory oversight and considering the conclusions of the Topical Peer Review it can be concluded that the ageing management of the Licensees satisfies the national regulations, and by this also the international requirements and recommendations. The ageing management of the Licensees is part of the activities aimed at maintaining the technical conditions of the systems, structures and components, in practice ageing management is the key of the coordinated implementation of the programmes. The quality assurance system of the Licensees provides that ageing management as a separate process is regularly reviewed, evaluated, the related experiences are collected and fed back.

Current review, thank to the systematic activity of the Licensees and the regulatory practice has not revealed any new, ageing management related deviation or place for improvement. The recognized ageing processes are managed by the Licensees and they are prepared to detect any unanticipated or new ageing mechanisms and to take into account the experiences of other facilities.

Paks NPP

In the service life extension process of the Paks NPP units individual comprehensive ageing management reviews have taken place. In the licensing, the efficiency of the ageing management programme had to be described regarding the SSCs out of the scope of service life extension, while within the scope (passive, long lived components) a comprehensive ageing review had to be performed by the Licensee. The ageing review aimed at demonstration of operability of the SSCs for the extended service life consisted of two parts: review of the ageing management programmes and validation of the time limited ageing analysis. As a result of the process, the activity was further improved on the basis of the deviations or places of improvements revealed partly by the Licensee itself partly by the HAEA. Based on the granted unit-level service life extension licences the compliance of the ageing management activity was accepted by the HAEA.

The above conclusions regarding the NPP are supported by the recently performed international reviews and comparisons. Namely the pre-SALTO and full scope SALTO missions implemented in Paks NPP (altogether 7 times between 2005 and 2011), the fulfilment of the recommendations and suggestions of which were confirmed by the follow-up missions. On the other hand in the

IAEA IGALL programme Paks NPP is a very active member and so it can study and utilize the experiences of others from the first hand to improve its own programme.

Within ageing management a good example of using the external experience that is mentioned in Section 02.3.2., but is out of the scope of this review, is the issue of modification of the MCPs in Paks NPP. Based on the Russian vendor recommendations, during the first period of the operation, the modification of the impeller/shaft took place according to the repair/replacement programme. The AMP, however, identified different damages, regarding which it turned out that operating experience and the damage process are related and that repair/replacement of the guide wheels and the compression heads became necessary.

Budapest Research Reactor

The ageing management programme of the Budapest Research Reactor was launched in 2005 based on IAEA recommendations, the NSC provisions and the results of the first PSR. During the 12 years of the programme irrespective of staff replacement there were no essential changes in the ageing management organization. In conclusion one of the major issues in this area is the ageing of the staff. It is more and more difficult for the research reactor to find and preserve on the long term the qualified, experienced work force. Considering that the induction training for certain jobs lasts for 3-5 years these issues needs immediate intervention from the management of the operator.

Until now, the availability of the SSCs did not decrease with two exceptions. The degradation processes could be timely identified and arrested or well mitigated by means of the decided actions. The mentioned two cases are the degradation of the main gate valves seals (this was not in the scope of the TPR) and the corrosion of the secondary carbon steel pipelines that needed longer repair times.

The residual lifetime of the SSCs in the scope of the ageing management programme is minimum 10 years, and certain components were replaced in spite of being proved to further operate (e.g. batteries).

Inspection of the condition of electric cables takes place according to the ageing management programme, their condition is appropriate.

Inspection of the reactor vessel is commensurate with the level of this age. There are no signs of flaws or corrosion.

Ageing management programme of the concealed pipelines were enhanced based on the experiences.

The age of the buildings approaches 60 years. Renewal of the building engineering systems is continuous, but a decision on service life extension would require a larger scale modernization, sealing against rain water and ground water, furthermore the face of the building would should be comprehensively renewed and modernized.

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List of Abbreviations

Abbreviation	Meaning
ABOS	Nuclear power plant safety classification
ACI	American Concrete Institute
AM	Ageing Management
AMP	Ageing Management Programme
AS6	IT administration system
ASME	American Society of Mechanical Engineers
ASME BPVC	ASME Boiler and Pressure Vessel Code
ASME BPVC III.	Section III: Rules for Construction of Nuclear Facility Components, Division 1
ASME BPVC XI.	Section XI: Rules for In-Service Inspection of Nuclear Power Plant Components, Division 1
BAMP	Building Ageing Management Programme
BRR	Budapest Research Reactor
BUTE	Budapest University of Technology and Economics
CANDU	CANada Deuterium Uranium
CG	Commodity group (= group consisting of similarly managed elements)
CLASS 1, 2, 3	1, 2. or 3. Strength classification; such components must comply with the requirements of ASME BPV Section III, Div 1, Subsec. NB, NC, and ND
COMSY	Condition Oriented ageing and plant life Monitoring System

Abbreviation	Meaning
CRDM	Control Rod Drive Mechanism
CUF	Cumulative Usage Factor
DAMP	Degradation oriented Ageing Management Programme
DBE	Design Base Earthquake
EN	European Norm
ENIQ	European Network of Inspection and Qualification
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
ETV	ERŐTERV Energy Design and Contracting Company
ETV-ERŐTERV Zrt.	Erőterv Energy Design and Contracting Private Limited Company
GALL	Generic Aging Lessons Learned
HAEA	Hungarian Atomic Energy Authority
HAS	Hungarian Academy of Sciences
HAS CER	Hungarian Academy of Sciences, Centre for Energy Research
HE-FM nozzle	Temperature Monitoring and Flux Measurement nozzle
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IEC	International Electrotechnical ComissionCommission
IEEE	Institute of Electrical and Electronics Engineers
IGALL	International Generic Ageing Lesson Learnt
ISI	In-Service Inspection
ISIP	In-Service Inspection Programme
KFKI	Central Research Institute for Physics
LF	Lead Factor
LOCA	Loss of Coolant Accident
MCP	Main Circulating Pump
MEM	Maintenance Effectiveness Monitoring
MIC	Microbiologically Influenced Corrosion
MSZ	Hungarian Standard
MVM Paks NPP Ltd.	Hungarian Electrical Works Paks Nuclear Power Plant Ltd.
NAR	National Assessment Report
ND	Nominal Diameter
NDT	Non Destructive Testing
NRC	Nuclear Regulatory Commission

Abbreviation	Meaning
NSC	Nuclear Safety Code
NUREG	Nuclear Regulatory Guide
OAMP	Overall Ageing Management Programme
OI	Operation Instructions
OLC	Operational Limits and Conditions
OOR	Organizational and Operational Rule
OWTS	Oscillating Wave Test System
Paks NPP	Paks Nuclear Power Plant
Paks NPP WENRA AMR	Ageing management report of the Paks NPP prepared according to the WENRA TPR technical specification
PSR	Periodic Safety Review
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
R&D	Research & Development
RPV	Reactor Pressure Vessel
SALTO	Safety Aspects of Long Term Operation (extrabudgetary programme of the IAEA)
SAM	Severe Accident Management
SAMP	Component Specific Ageing Management Programme
SR	Safety related environmental characteristics
SSCs	Systems, Structures and Components
TBE	Equipment Lifecycle management processes ID assigned to the Production subsystem within the plant's internal management system
TI	Technology Instruction
TLAA	Time Limited Ageing Analysis
TLAS	Two Level Acceptance Standards
US	Ultrasonic
US NRC	United States Nuclear Regulatory Commission
VVER	Водо-водяной энергетический реактор (water-water energetic reactor)
WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulators Association

11. Attachments

1.1 ANNEX 02.3.1.3-1: AGING MANAGEMENT COMMODITY GROUPS OF MECHANICAL COMPONENTS AND THE RELATED SAMP, SCG IDs

Compo-nent	SAMP ID	SCG ID	Medium1	Medium2	Material
Priority main equipment	SAMP-001: Reactor pressure vessels	-	-	-	-
	SAMP-002: Reactor pressure vessel internals	-	-	-	-
	SAMP-005: Pressurizer	-	-	-	-
	SAMP-006: Steam generators	-	-	-	-
	SAMP-009: Main circulating pipes	-	-	-	-
	SAMP-007: Main gate valves	-	-	-	-
	SAMP-008: Main circulating pumps	-	-	-	-
Heat exchangers	W-SAMP-01	W154	Primary circuit water	Danube water	Mixture (Carbon steel, non-corrosion-resistant steels / corrosion resistant steels)
	W-SAMP-02	W132	Primary circuit water	Danube water	Corrosion resistant steels
		W232	Primary circuit steam	Danube water	Corrosion resistant steels
		W632	Other contaminated solution	Treated water	Corrosion resistant steels
	W-SAMP-03	W112	Primary circuit water	Primary circuit water	Corrosion resistant steels
	W-SAMP-05	W152	Primary circuit water	Danube water	Corrosion resistant steels
		W252	Primary circuit steam	Danube water	Corrosion resistant steels
	W-SAMP-07	W585	Danube water	Gas	Mixture (Corrosion resistant steels/Other alloys)
	W-SAMP-08	W582	Danube water	Gas	Corrosion resistant steels

Compo-nent	SAMP ID	SCG ID	Medium1	Medium2	Material
	W-SAMP-10	W354	Treated water	Danube water	Mixture (Carbon steel, non-corrosion-resistant steels / corrosion resistant steels)
	W-SAMP-11	W351	Treated water	Danube water	Carbon steel, non-corrosion-resistant steels
	W-SAMP-12	W352	Treated water	Danube water	Corrosion resistant steels
		W1052	Steam-gas mixture	Danube water	Corrosion resistant steels
	W-SAMP-13	W381	Treated water	Gas	Carbon steel, non-corrosion-resistant steels
	W-SAMP-16	W571	Danube water	Oil	Carbon steel, non-corrosion-resistant steels
	W-SAMP-17	W801	Gas	-	Carbon steel, non-corrosion-resistant steels
	W-SAMP-18	W754	Oil	Danube water	Mixture (Carbon steels, non-corrosion resistant steels / Corrosion resistant steels)
		W756	Oil	Danube water	Mixture (Carbon steels, non-corrosion-resistant steels /Other alloys)
		W757	Oil	Danube water	Mixture (Carbon steels, non-corrosion-resistant steels / corrosion-resistant steels / Other alloys)
	W-SAMP-19	W334	Treated water	Treated water	Mixture (Carbon steel, non-corrosion-resistant steels / corrosion resistant steels)
		W344	Treated water	Water steam	Mixture (Carbon steel, non-corrosion-resistant steels / corrosion resistant steels)
	W-SAMP-20	W586	Danube water	Gas	Mixture (Carbon steels, non-corrosion-resistant steels/Other alloys)
	W-SAMP-21	W1032	Treated water steam gas	Treated water	Corrosion resistant steels
	W-SAMP-22	W386	Treated water	Gas	Mixture (Carbon steels, non-corrosion-resistant steels /Other alloys)
	W-SAMP-23	W536	Danube water	Treated water	Mixture (Carbon steels, non-corrosion-resistant steels/Other alloys)
	W-SAMP-24	W376	Treated water	Oil	Carbon steels, non-corrosion-resistant steels, Other alloys

Compo-nent	SAMP ID	SCG ID	Medium1	Medium2	Material
	W-SAMP-25	W755	Oil	Danube water	Mixture (Corrosion-resistant steels /Other alloys)
Vessels	B-SAMP-01	B501	Danube water	-	Carbon steel, non-corrosion-resistant steels
	B-SAMP-03	B602	Other contaminated solution	-	Corrosion resistant steels
		B642	Other contaminated solution	Water steam	Corrosion resistant steels
		B902	Acid/base	-	Corrosion resistant steels
	B-SAMP-04	B801	Gas	-	Carbon steel, non-corrosion-resistant steels
	B-SAMP-05	B802	Gas	-	Corrosion resistant steels
	B-SAMP-06	B301	Treated water	-	Carbon steel, non-corrosion-resistant steels
		B341	Treated water	Water steam	Carbon steel, non-corrosion-resistant steels
	B-SAMP-07	B302	Treated water	-	Corrosion resistant steels
	B-SAMP-08	B701	Oil	-	Carbon steel, non-corrosion-resistant steels
		B781	Oil	Gas	Carbon steel, non-corrosion-resistant steels
	B-SAMP-09	B102	Primary circuit water	-	Corrosion resistant steels
		B112	Primary circuit water	Primary circuit water	Corrosion resistant steels
		B162	Primary circuit water	Other contaminated solution	Corrosion resistant steels
		B122	Primary circuit water	Primary circuit steam	Corrosion resistant steels
		B282	Primary circuit steam	Gas	Corrosion resistant steels
	B-SAMP-10	B344	Treated water	Water steam	Mixture (Carbon steel, non-corrosion-resistant steels / corrosion resistant steels)
	B-SAMP-11	B702	Oil	-	Corrosion resistant steels
	B-SAMP-12	B861	Gas	Other dirt	Carbon steel, non-corrosion-resistant steels
Pipeline and pipeline	Z-SAMP-01	Z504	Danube water	-	Mixture (Carbon steel, non-corrosion-resistant steels / corrosion resistant steels)

Compo-nent	SAMP ID	SCG ID	Medium1	Medium2	Material
	Z-SAMP-02	Z501	Danube water	-	Carbon steel, non-corrosion-resistant steels
		Z531	Danube water	Treated water	Carbon steel, non-corrosion-resistant steels
		Z561	Danube water	Other contaminated solution	Carbon steel, non-corrosion-resistant steels
	Z-SAMP-03	Z502	Danube water	-	Corrosion resistant steels
		Z532	Danube water	Treated water	Corrosion resistant steels
	Z-SAMP-04	Z102	Primary circuit water	-	Corrosion resistant steels
		Z132	Primary circuit water	Treated water	Corrosion resistant steels
		Z142	Primary circuit water	Water steam	Corrosion resistant steels
		Z152	Primary circuit water	Danube water	Corrosion resistant steels
		Z162	Primary circuit water	Other contaminated solution	Corrosion resistant steels
		Z182	Primary circuit water	Gas	Corrosion resistant steels
		Z192	Primary circuit water	Acid/Base	Corrosion resistant steels
		Z202	Primary circuit steam	-	Corrosion resistant steels
		Z282	Primary circuit steam	Gas	Corrosion resistant steels
		Z692	Other contaminated solution	Acid/base	Corrosion resistant steels
		Z902	Acid/base	-	Corrosion resistant steels
	Z-SAMP-05	Z602	Other contaminated solution	-	Corrosion resistant steels
		Z632	Other contaminated solution	Treated water	Corrosion resistant steels
		Z672	Other contaminated solution	Oil	Corrosion resistant steels

Compo-nent	SAMP ID	SCG ID	Medium1	Medium2	Material
		Z682	Other contaminated solution	Gas	Corrosion resistant steels
	Z-SAMP-06	Z701	Oil	-	Carbon steel, non-corrosion-resistant steels
	Z-SAMP-07	Z801	Gas	-	Carbon steel, non-corrosion-resistant steels
	Z-SAMP-08	Z802	Gas	-	Corrosion resistant steels
	Z-SAMP-09	Z304	Treated water	-	Mixture (Carbon steel, non-corrosion-resistant steels / corrosion resistant steels)
		Z344	Treated water	Water steam	Mixture (Carbon steel, non-corrosion-resistant steels / corrosion resistant steels)
		Z404	Water steam	-	Mixture (Carbon steel, non-corrosion-resistant steels / corrosion resistant steels)
	Z-SAMP-10	Z301	Treated water	-	Carbon steel, non-corrosion-resistant steels
		Z341	Treated water	Water steam	Carbon steel, non-corrosion-resistant steels
		Z401	Water steam	-	Carbon steel, non-corrosion-resistant steels
	Z-SAMP-11	Z302	Treated water	-	Corrosion resistant steels
		Z342	Treated water	Water steam	Corrosion resistant steels
		Z382	Treated water	Gas	Corrosion resistant steels
		Z402	Water steam	-	Corrosion resistant steels
		Z482	Water steam	Gas	Corrosion resistant steels
	Z-SAMP-12	Z601	Other contaminated solutions	-	Carbon steel, non-corrosion-resistant steels
	Z-SAMP-13	Z804	Gas	-	Mixture (Carbon steel, non-corrosion-resistant steels / corrosion resistant steels)
	Z-SAMP-14	Z803	Gas	-	Other alloys
	Z-SAMP-15	Z702	Oil	-	Corrosion resistant steels
	Z-SAMP-16	Z704	Oil	-	Mixture (Carbon steel, non-corrosion-resistant steels / corrosion resistant steels)
Concealed pipel	Z-SAMP-17	TG system, corrosion resistant steel, partially concealed pipework encased in concrete			

<table>
<tr><th>Compo-nent</th><th>SAMP ID</th><th>SCG ID</th><th>Medium1</th><th>Medium2</th><th>Material</th></tr>
<tr><td rowspan="7"></td><td>Z-SAMP-18</td><td colspan="4">Partially concealed carbon steel pipework in pipe tunnels, buried in soil, encased in concrete, covered in trenches used to transfer Danube water</td></tr>
<tr><td>Z-SAMP-19</td><td colspan="4">Partially concealed corrosion resistant steel pipework encased in concrete used to transfer presumably radioactive primary circuit water, other contaminated solutions</td></tr>
<tr><td>Z-SAMP-20</td><td colspan="4">Partially concealed corrosion resistant steel pipework encased in concrete used to transfer presumably non-radioactive primary circuit water, other contaminated solutions</td></tr>
<tr><td>Z-SAMP-21</td><td colspan="4">Partially concealed pipework in trenches, buried in soil, made of carbon steel, non-corrosion resistant steel used to transfer diesel oil</td></tr>
<tr><td>Z-SAMP-22</td><td colspan="4">Concealed pipework (ducts) partially encased in concrete made of carbon steel, non-corrosion resistant steel used to transfer air</td></tr>
<tr><td>Z-SAMP-23</td><td colspan="4">Concealed pipework (ducts), (partially) encased in concrete made of corrosion resistant steel used to transfer air</td></tr>
<tr><td>Z-SAMP-24</td><td colspan="4">Partially concealed pipework in trenches, pipe tunnel, made of carbon steel, non-corrosion resistant steel used to transfer treated water</td></tr>
<tr><td rowspan="17">Valves</td><td>S-SAMP-01</td><td>S501</td><td>Danube water</td><td>-</td><td>Carbon steel, non-corrosion-resistant steels</td></tr>
<tr><td>S-SAMP-02</td><td>S502</td><td>Danube water</td><td>-</td><td>Corrosion-resistant steels</td></tr>
<tr><td rowspan="2">S-SAMP-03</td><td>S601</td><td>Other contaminated solution</td><td>-</td><td>Carbon steel, non-corrosion-resistant steels</td></tr>
<tr><td>S901</td><td>Acid/base</td><td>-</td><td>Carbon steel, non-corrosion-resistant steels</td></tr>
<tr><td rowspan="3">S-SAMP-04</td><td>S602</td><td>Other contaminated solution</td><td>-</td><td>Corrosion resistant steels</td></tr>
<tr><td>S692</td><td>Other contaminated solution</td><td>Acid/base</td><td>Corrosion resistant steels</td></tr>
<tr><td>S902</td><td>Acid/base</td><td>-</td><td>Corrosion resistant steels</td></tr>
<tr><td rowspan="2">S-SAMP-05</td><td>S801</td><td>Gas</td><td>-</td><td>Carbon steel, non-corrosion-resistant steels</td></tr>
<tr><td>S881</td><td>Gas</td><td>Gas</td><td>Carbon steel, non-corrosion-resistant steels</td></tr>
<tr><td rowspan="2">S-SAMP-06</td><td>S883</td><td>Gas</td><td>Gas</td><td>Other alloys</td></tr>
<tr><td>S803</td><td>Gas</td><td>-</td><td>Other alloys</td></tr>
<tr><td rowspan="2">S-SAMP-07</td><td>S802</td><td>Gas</td><td>-</td><td>Corrosion resistant steels</td></tr>
<tr><td>S882</td><td>Gas</td><td>Gas</td><td>Corrosion resistant steels</td></tr>
<tr><td rowspan="2">S-SAMP-08</td><td>S301</td><td>Treated water</td><td>-</td><td>Carbon steel, non-corrosion-resistant steels</td></tr>
<tr><td>S341</td><td>Treated water</td><td>Water steam</td><td>Carbon steel, non-corrosion-resistant steels</td></tr>
</table>

Compo-nent	SAMP ID	SCG ID	Medium1	Medium2	Material
		S401	Water steam	-	Carbon steel, non-corrosion-resistant steels
	S-SAMP-09	S302	Treated water	-	Corrosion-resistant steels
		S342	Treated water	Water steam	Corrosion-resistant steels
		S402	Water steam	-	Corrosion-resistant steels
	S-SAMP-10	S701	Oil	-	Carbon steel, non-corrosion-resistant steels
		S703	Oil	-	Other alloys
	S-SAMP-11	S702	Oil	-	Corrosion resistant steels
	S-SAMP-12	S102	Primary circuit water	-	Corrosion resistant steels
		S162	Primary circuit water	Other contaminated solution	Corrosion resistant steels
		S202	Primary circuit steam	-	Corrosion resistant steels
	S-SAMP-13	S503	Danube water	-	Other alloys
	S-SAMP-15	S861	Gas	Other dirt	Carbon steel, non-corrosion-resistant steels
	S-SAMP-16	S303	Treated water	-	Other alloys
		S403	Water steam	-	Other alloys
Pumps	D-SAMP-01	D501	Danube water	-	Carbon steel, non-corrosion-resistant steels
	D-SAMP-03	D601	Other dirt	-	Carbon steel, non-corrosion-resistant steels
	D-SAMP-04	D602	Other contaminated solution	-	Corrosion resistant steels
		D692	Other contaminated solution	Acid/base	Corrosion resistant steels
		D902	Acid/base	-	Corrosion resistant steels
	D-SAMP-05	D102	Primary circuit water	-	Corrosion resistant steels
	D-SAMP-06	D301	Treated water	-	Carbon steel, non-corrosion-resistant steels
	D-SAMP-07	D302	Treated water	-	Corrosion resistant steels
	D-SAMP-08	D701	Oil	-	Carbon steel, non-corrosion-resistant steels

Compo-nent	SAMP ID	SCG ID	Medium1	Medium2	Material
Filters	N-SAMP-01	N102	Primary circuit water	-	Corrosion resistant steels
		N162	Primary circuit water	Other contaminated solution	Corrosion resistant steels
		N182	Primary circuit water	Gas	Corrosion resistant steels
		N192	Primary circuit water	Acid/base	Corrosion resistant steels
		N282	Primary circuit steam	Gas	Corrosion resistant steels
	N-SAMP-02	N602	Other contaminated solution	-	Corrosion resistant steels
		N632	Other contaminated solution	Treated water	Corrosion resistant steels
	N-SAMP-03	N801	Gas	-	Carbon steel, non-corrosion-resistant steels
		N806	Gas	-	Mixture (Carbon steels, non-corrosion-resistant steels /Other alloys)
	N-SAMP-04	N802	Gas	-	Corrosion resistant steels
	N-SAMP-05	N302	Treated water	-	Corrosion resistant steels
		N402	Water steam	-	Corrosion resistant steels
	N-SAMP-06	N701	Oil	-	Carbon steel, non-corrosion-resistant steels
		N706	Oil	-	Mixture (Carbon steels, non-corrosion-resistant steels/Other alloys)
	N-SAMP-07	N803	Gas	-	Other alloys
		N805	Gas	-	Mixture (corrosion resistant steels /Other alloys)
Individually managed special structures	Evaporators SAMP	-	-	-	-
	Bubbling condenser trays SAMP	-	-	-	-
	Condenser-degasser SAMP	-	-	-	-
	Make-up water and boron control degasser SAMP	-	-	-	-
	Air-conditioner heat exchanger SAMP	-	-	-	-

Compo-nent	SAMP ID	SCG ID	Medium1	Medium2	Material
	Shafts-SAMP	-	-	-	-
	Hermetic penetrations -SAMP	-	-	-	-
	Hermetic openings -SAMP	-	-	-	-
	Hydro accumulator tanks-SAMP	-	-	-	-
	Robust structures SAMP	-	-	-	-
	Fain-coil-SAMP	-	-	-	-
	Plastic components SAMP	-	-	-	-
	Turbine separator and overheating SAMP	-	-	-	-
	Gas blowers-SAMP	-	-	-	-
	Compressor-SAMP	-	-	-	-
Fans	DV-SAMP-01	DV801	Gas	-	Carbon steel, non-corrosion-resistant steels
	DV-SAMP-02	DV802	Gas	-	Corrosion resistant steels
		DV803	Gas	-	Other alloys
Impulse tubes	ZI-SAMP-002	-	-	-	-
	ZI-SAMP-001	Z102/a	Primary circuit water	-	Corrosion resistant steels
		Z102/b	Primary circuit water	-	Corrosion resistant steels
		Z282/a	Primary circuit steam	Gas	Corrosion resistant steels

1.2 ANNEX 02.3.1.3-2: AGEING MANAGEMENT GROUPS FOR CIVIL STRUCTURES AND THE ASSOCIATED STRUCTURE AMPs

Main group, based on material and structure	Structure component group	Name of structure AMP
Reinforced concrete structures	Foundations (plate bases, band bases, pillars), retaining walls, below-grade reinforced structures	BAMP-A-002 Foundations AMP BAMP-A-008 Monitoring of building settlement AMP BAMP-A-003 Monitoring of base points AMP
	Superstructures (including machine bases)	BAMP-A-021 Reinforced concrete superstructures AMP BAMP-A-005 Surveillance of boric acid corrosion of reinforced concrete structures AMP BAMP A-011 Degradation of machine base due to fatigue AMP BAMP-A-015 Concrete structures exposed to elevated temperature AMP
	Prefabricated reinforced concrete walls, slabs and columns of centrifuged concrete	BAMP-A-021 Reinforced concrete superstructures AMP BAMP-A-005 Surveillance of boric acid corrosion reinforced concrete structures AMP BAMP-A-015 Concrete structures exposed to elevated temperature AM
	Civil engineering hydraulics reinforced concretes (base plates, bottom plates, walls, cutoff walls, floors, pavements, pillars, head rails and mooring column)	BAMP-A-014 Civil engineering hydraulic utility concrete structures AMP BAMP-A-002 Foundations AMP
Steel structures	Carbon steel building and frame structures, slabs, various support structures, crane tracks, platforms	BAMP-A-001 Steel structures AMP BAMP-A-005 Surveillance of boric acid corrosion of reinforced concrete structures AMP
	Sheet metal claddings	BAMP-A-001 Steel structures AMP BAMP-A-005 Surveillance of boric acid corrosion of reinforced concrete structures AMP
	Hermetic lining	BAMP-A-012 Hermetic lining AMP BAMP-A-013 Leak-tightness of hermetic compartments AMP BAMP-A-005 Surveillance of boric acid corrosion of reinforced concrete structures AMP
	Stainless steel cladding	BAMP-A-018 Stainless steel plates AMP

Main group, based on material and structure	Structure component group	Name of structure AMP
	Civil engineering hydraulic steel structures	BAMP-A-023 Steel structures of hydraulic engineering AMP
Earthworks	Dikes	BAMP-A-010 Earthen embankments AMP
	Channel bottom and slope paving	BAMP-A-010 Earthen embankments AMP
Masonry walls	Masonry walls	BAMP-A-009 Masonry walls AMP
Coatings	Synthetic resin coatings, corrosion protection coatings	BAMP-A-004 Coatings AMP
Fire propagation barrier structures	Reinforced concrete walls, masonry walls	BAMP-A-020 Fire barrier structures AMP BAMP-A-021 Reinforced concrete superstructures AMP BAMP-A-005 Surveillance of boric acid corrosion of reinforced concrete structures AMP BAMP-A-015 BAMP-A-015 Concrete structures exposed to elevated temperature AMP BAMP-A-009 Masonry walls AMP
	Fire barrier doors, coatings and cladding, fire barrier structures made from steel and asbestos	BAMP-A-020 Fire barrier structures AMP
Doors and hatches	Biological protection and hermetic doors and hatches (also possessing fire barrier functions)	BAMP-A-016 Doors and hatches AMP BAMP-A-013 Leak-tightness of the hermetic compartment AMP BAMP-A-020 Fire barrier structures AMP
Seals, dilatation joints	Sealant of biological protection and hermetic doors and hatches	BAMP-A-019 Seals AMP
	Seals of fire barrier doors	BAMP-A-019 Seals AMP
	Other seals, strip seals for expansion joints	BAMP-A-019 Seals AMP
Main group of other commodity groups managed in unique programmes	Pipe and cable support structures	BAMP-A-006 Pipe and cable support structures AMP
	Electrical distributors, control and relay panels, local actuation boards	BAMP-A-022 Electrical distributors, control and relay tables, local actuation cabinet AMP

1.3 ANNEX 03.1.1-1: CABLE SAMPLE GROUPS

Cable groups of U>3 kV cables and 380 V < U < 3 kV cables

Groups indicated by the green background are U>3 kV cables	Commodity groups with white backgrounds are 380 V < U < 3 kV cables

No	Commodity group	Type group	SR no.	SR length (m)	SR23 no.	SR23 length (m)	Description of qualified state and their justification	Qualification
1	CG_(N)HXSHXÖ	(N)HXSHXÖ Sienopyr X	188	4258	164	3837	Ambient operating temperature up to a maximum of 60 °C. Some also have functions during emergency conditions and severe accident management (maximum temperature 160 °C). Artificial ageing Thermal operating simulation: 90 °C/425 days/41 years Gamma operating simulation 94 kGy, Gamma emergency: 110 kGy Document: LOCA-19/2009-3	Pass: 429/VNL Product certificate
2	CG_H07RN-F	GT-H07RN-F	56	410	44	290	Ambient operating temperature up to a maximum of 35 °C. Some also have functions during emergency conditions (maximum temperature 34 °C). The cable operates without radiation	Pass: 366/VNL Product certificate
3	CG_Hydrofirm	Hydrofirm_TKG	60	300	42	210	Ambient operating temperature up to a maximum of 60 °C. Some also have functions during emergency conditions and severe accident management (maximum temperature 130 °C). Artificial Ageing Thermal operating simulation: 90 °C/333 days/27 years Gamma operating simulation 2.37 kGy, Gamma emergency: 110 kGy Document: LOCA-04/2013-K14	Pass: 5029/VNL Product certificate
4	CG_JZ-600	JZ-600	171	2110	130	1825	Ambient operating temperature up to a maximum of 40 °C. Some also have functions during emergency conditions and severe accident management (maximum temperature 156 °C). Artificial Ageing Thermal operating simulation: 80 °C/134 days/37 years Gamma operating simulation: 84.8 kGy, Gamma emergency: 110 kGy Document: LOCA-19/2009-3	Pass: 367/VNL Product certificate
5	CG_KEFSZ	KEFSZ	5	265	4	95	Ambient operating temperature up to a maximum of 58 °C. Some also have functions during emergency conditions (maximum temperature 56 °C). Artificial Ageing Thermal operating simulation: 110 °C/249 days/50 years	Pass: V01-18907_C28-2011 Certificate

No	Commodity group	Type group	SR no.	SR length (m)	SR23 no.	SR23 length (m)	Description of qualified state and their justification	Qualification
							Gamma operating simulation - Document: 00000006V00003VMA	
6	CG_KMPEV	KMPEV	47	4 385	30	2 900	Ambient operating temperature up to a maximum of 35 °C. Some also have functions during emergency conditions (maximum temperature 93 °C). Successful operating type inspections following accelerated ageing, Document: 000006V00001VMA V01-24761_C11-2010 Successful inspections simulating emergency parameters later. Document: LOCA-19/2009-7	Pass: V01-24761_C11-2010, 412/VNL Product certificate
7	CG_KPO	KPO	267	14613	224	13257	Ambient operating temperature up to a maximum of 60 °C. Some also have functions during emergency conditions (maximum temperature 148 °C). <u>Artificial Ageing:</u> 21 years of ageing on cables removed from the operating environment after 29 years. Thermal operating simulation: 90 °C/218 days/21 years Gamma operating simulation 94.86 kGy, Gamma emergency: 110 kGy Document: LOCA-19/2009-9	Pass: 446/VNL Product certificate
8	CG_KUGV	KUGV	3	205	0	0	Ambient operating temperature up to a maximum of 30 °C. Do not have functions during emergency conditions. <u>Artificial Ageing</u> Thermal operating simulation: 65 °C/413 days/50 years Gamma operating simulation: - Document: 000006V00001VMA	Pass: V01-24761_C12-2011 Certificate
9	CG_KVV	KVVGE	255	19196	75	6875	Ambient operating temperature up to a maximum of 35 °C. Some also have functions during emergency conditions (maximum temperature 148 °C). <u>Artificial Ageing:</u> Thermal operating simulation:: 65 °C/647 days/50 years Gamma operating simulation: - Gamma emergency: - Document: 000006V00002VMA	Pass: V01-18907_C30-2011 Certificate
10	CG_KVV_E412_Function-analysis	KVVGE u.a.	13	1500	13	1500	Ambient operating temperature up to a maximum of 35 °C. The cables also have functions during emergency conditions (maximum temperature 219 °C) The operating type inspections have been successful. During emergency inspections however reaching 190 °C (2020 sec) the cable broke. According to the function analysis (document 000000A00218 ERA) the cables must be operational within the first 160 sec during the 2020 sec emergency testing,	Pass: V01-18907_C29-2011. Certificate

No	Commodity group	Type group	SR no.	SR length (m)	SR23 no.	SR23 length (m)	Description of qualified state and their justification	Qualification
							therefore the cables are operational for the period of time needed to perform their emergency functions. Artificial Ageing: Thermal operating simulation: 65 °C/413 days/50 years Gamma operating simulation: - Gamma emergency: - Document: 000006V00002VMA	
11	CG_KVV_ ÖRK	KVVGE u.a	9	515	0	0	The concerned cables operate normally in ambient temperature up to a maximum of 40 °C. This is higher than for cables belonging to the core-commodity group (CG_KVV, therefore the qualification of the operating condition of cables belonging to the core-commodity group cannot be applied to them in the same way. They do not have functions during emergency. Sample cables designated from the type cables are monitored according to the V-SAMP-07 special ageing management program.	Pass: managed in V-SAMP-07
12	CG_KVV_ ÖRK_Function-analysis	KVVGE	13	610	13	610	The concerned cables operate normally in ambient temperature up to a maximum of 40 °C. This is higher than for cables belonging to the core-commodity group (CG_KVV). Their ageing is therefore faster. Therefore, the qualification of the operating condition of cables belonging to the core-commodity group cannot be applied to them in the same way. They do not have functions during emergency. Sample cables assigned from the type cables are monitored according to the V-SAMP-07 special ageing management program.	Pass: managed in V-SAMP-07
13	CG_Li9Y	Li9Y11Y	11	31	0	0	New cables operating in mild environment since 2010, whose operating qualification is not required. Ambient operating temperature is max. 30 °C. The operating test following artificical ageing for the duration of the service life extension were successful. Artificial Ageing Thermal operating simulation: 90 °C/14 days/28 years Document: LOCA-19/2009-3	Pass: 13/VNL Product Certificate
14	CG_MD/90	MD/90	56	1255	0	0	Ambient operating temperature up to a maximum of 35 °C. Do not have functions during emergency conditions. Artificial Ageing Thermal operating simulation: 90 °C/91 days/50 years Gamma operating simulation: - Document: 000006V00003VMA	Pass: V01-18907_C31-2011 Certificate

No	Commodity group	Type group	SR no.	SR length (m)	SR23 no.	SR23 length (m)	Description of qualified state and their justification	Qualification
15	CG_N2XS	N2XS	32	17058	12	9 000	Ambient operating temperature up to a maximum of 49 °C. Some also have functions during emergency conditions (maximum temperature 93 °C). Artificial Ageing: Thermal operating simulation: 90 °C/151 days/32 years Gamma operating simulation: 25.8 Gy Document: LOCA-19/2009-3 Emergency test is not required, because the cables are laid in independent, 90 minute fire rating concrete trenches.	Pass 375/VNL Product Certificate
16	CG_NHXH E30/E90	NHXH_E30/E90	122	14182	103	13415	Ambient operating temperature up to a maximum of 60 °C. Some also have functions during emergency conditions (maximum temperature 217 °C). Artificial Ageing: Thermal operating simulation: 90 °C/425 days/41 years Gamma operating simulation: 94 kGy Gamma emergency: 110 kGy Document: LOCA-19/2009-3	Pass 363/VNL Product Certificate
17	CG_NHXH X	NHXHX Sienopyr X	18	2 457	18	2 457	Ambient operating temperature up to a maximum of 55 °C. Some also have functions during emergency conditions (maximum temperature 217 °C). Artificial Ageing: Thermal operating simulation: 90 °C/483 days/40 years Gamma operating simulation: 92 kGy Gamma emergency: 110 kGy Document: LOCA-19/2009-3	Pass: 362/VNL Product Certificate
18	CG_NSSH	NSSHöu	1	90	0	0	Ambient operating temperature up to a maximum of 36 °C. Do not have functions during emergency conditions. Artificial Ageing: Thermal operating simulation: 80 °C/56 days/38 years Gamma operating simulation: 86 Gy Document: LOCA-19/2009-4	Pass: 384/VNL Product Certificate
19	CG_NTSWöu-J	NTSWöu-J	54	1815	0	0	Ambient operating temperature up to a maximum of 30 °C. Do not have functions during emergency conditions. Artificial Ageing: Thermal operating simulation: 80 °C/55 days/43 years Document: LOCA-19/2009-3	Pass 368/VNL Product Certificate
20	CG_NU-HXH	NU-HXH	534	26166	516	25486	Ambient operating temperature up to a maximum of 60 °C. Some also have functions during emergency conditions and severe accident management (maximum temperature 160 °C).	Pass: 400/VNL Product Certificate

No	Commodity group	Type group	SR no.	SR length (m)	SR23 no.	SR23 length (m)	Description of qualified state and their justification	Qualification
							Artificial Ageing Thermal operating simulation: 110 °C/35 days/23 years Gamma operating simulation 52.6 kGy, Gamma emergency: 33 kGy Document: LOCA-19/2009-5	
21	CG_NYCWY	NYCWY	238	11009	52	4462	Ambient operating temperature up to a maximum of 36 °C. Some also have functions during emergency conditions and severe accident management (maximum temperature 130 °C). Artificial Ageing Thermal operating simulation: 70 °C/251 days/37 years Gamma operating simulation 83.4 kGy Document: LOCA-19/2009-3	Pass: 374/VNL Product Certificate
22	CG_NYY	NYY	971	62538	167	15522	Ambient operating temperature up to a maximum of 40 °C. Some also have functions during emergency conditions and severe accident management (maximum temperature 163 °C). Artificial Ageing Thermal operating simulation: 70 °C/502 days/41 years Gamma emergency: 92.4 kGy Document: LOCA-19/2009-3	Pass: 365/VNL Product Certificate
23	CG_SiHF	SiHF	189	1544	154	1024	Ambient operating temperature up to a maximum of 60 °C. Some also have functions during emergency conditions (maximum temperature 219 °C). Artificial Ageing Thermal operating simulation: 10 °C/32 days/23 years Gamma operating simulation 52.6 kGy, Gamma emergency: 33 kGy Document: LOCA-19/2009-6	Pass: 369/VNL Product Certificate
24	CG_SiHF_SB	SiHF	70	2395	70	2395	Ambient operating temperature up to a maximum of 60 °C. Some also have functions during emergency conditions and severe accident management (maximum temperature 160 °C). Artificial Ageing Thermal operating simulation: 10 °C/32 days/23 years Gamma operating simulation 52.6 kGy, Gamma emergency: 33 kGy Document: LOCA-19/2009-6	Pass 401/VNL Product Certificate
25	CG_SZAM	SZAM	860	62852	423	38727	Ambient operating temperature up to a maximum of 40 °C. Some also have functions during emergency conditions and severe accident management (maximum temperature 219 °C).	Pass V01-18907_C34-2011 Certificate

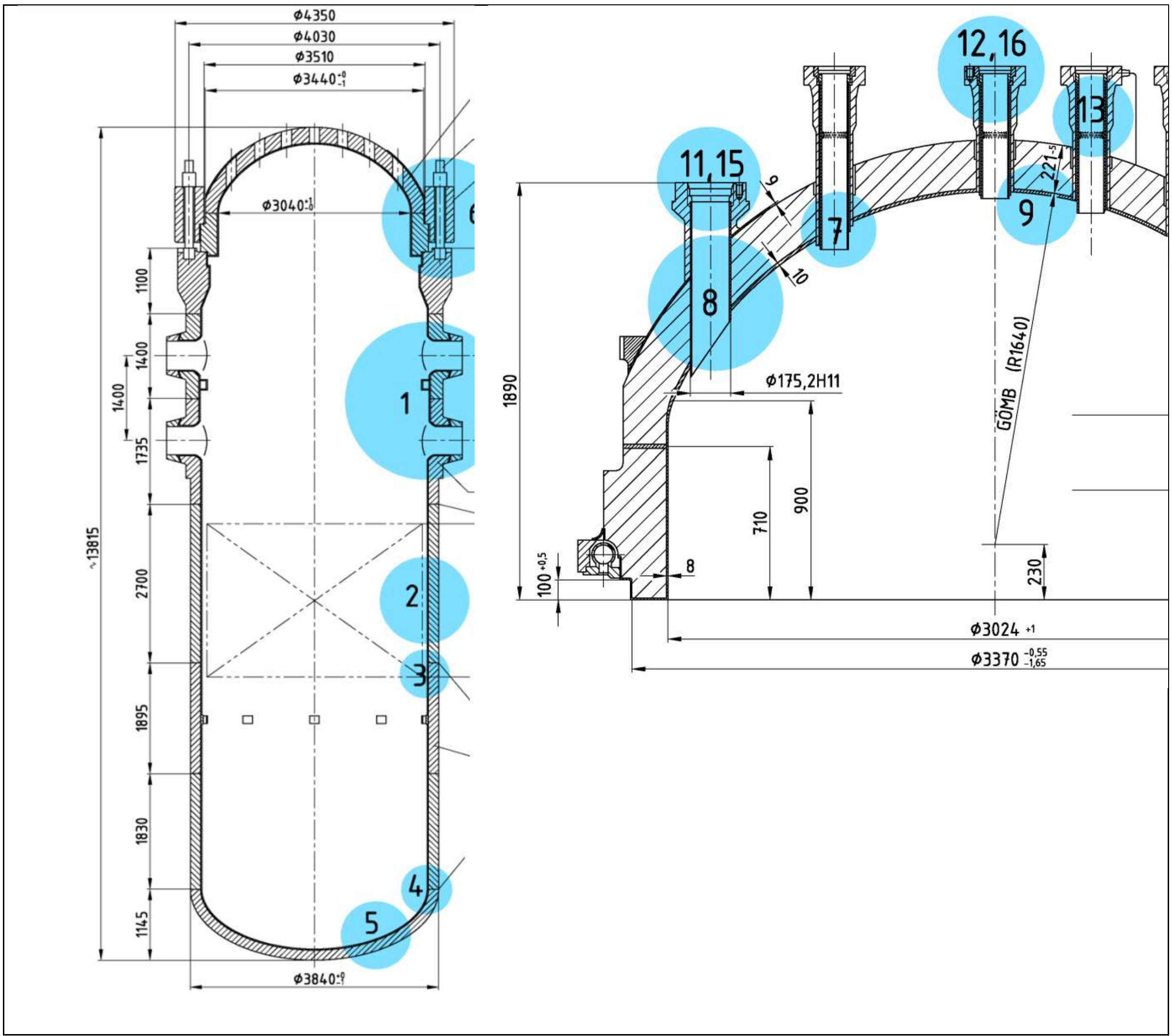
No	Commodity group	Type group	SR no.	SR length (m)	SR23 no.	SR23 length (m)	Description of qualified state and their justification	Qualification
							Artificial Ageing Thermal operating simulation: 70 °C/751 days/50 years Gamma operating simulation: - , Gamma emergency: - Document: 000006V00002VMA	
26	CG_SZAMK6	SZAMK-6	60	6495	14	2255	Ambient operating temperature up to a maximum of 35 °C. Some also have functions during emergency conditions and severe accident management (maximum temperature 100 °C). Artificial Ageing Thermal operating simulation: 70 °C/89 days/10 years Document: LOCA-220-2007-6	Pass 345/VNL Product Certificate
27	CG_SZRMK	SZRMK	10148	830740	3355	368763	Ambient operating temperature up to a maximum of 40 °C. Some also have functions during emergency conditions and severe accident management (maximum temperature 219 °C). Artificial Ageing Thermal operating simulation: 70 °C/748 days/50 years Gamma operating simulation: - , Gamma emergency: - Document: 000006V00002VMA	Pass V01-18907_C35-2011 Certificate
28	CG_MGPI	MGPI	28	1248	28	1248	Ambient operating temperature up to a maximum of 60 °C. The cables also also have functions during emergency conditions (maximum temperature 160 °C). Therefore their emergency qualification is also required. Their compliance is verified by the manufacturer's documentation of the qualifying tests carried out together with the connected equipment. Artificial Ageing Thermal operating simulation: Maximum 135 °C certified for 40 years Gamma operating simulation: - TID $2*10^6$ Gy Document: MGPI_adatlap / ETS1/KG50SEC_1.	Pass: MGPI_adatlap / ETS1/KG50SEC_1.
29	CG_FIREWALL	FIREWALL	64	266	64	266	Ambient operating temperature up to a maximum of 46 °C. The cables also have functions during emergency conditions (maximum temperature 136 °C). Therefore their emergency qualification is also required. Their compliance is verified by the manufacturer's documentation of the qualifying tests carried out together with the connected equipment obtained prior to their installation. Artificial Ageing Thermal operating simulation: 150 °C/941 days (40 year maximum 90 °C)	Pass: Firewall_III_QR-5804_Qualification_Report

No	Commodity group	Type group	SR no.	SR length (m)	SR23 no.	SR23 length (m)	Description of qualified state and their justification	Qualification
							Gamma operating simulation: - TDR 200.97 Mrad, Thermal/gamma emergency: LOCA/DBE profile of IEEE-323-1974, Document: Firewall_III_ QR-5804_Qualification_Report.	
30	CG_NTSCgEWöu	NTSCgEwöu	12	110	0	0	Ambient temperature up to a maximum of 30 °C. Do not have functions during emergency conditions. Artificial Ageing: Thermal operating simulation: maximum 30 °C certified for 48 years Document: LOCA-19/2009-3	Pass: 370/VNL Product Certificate

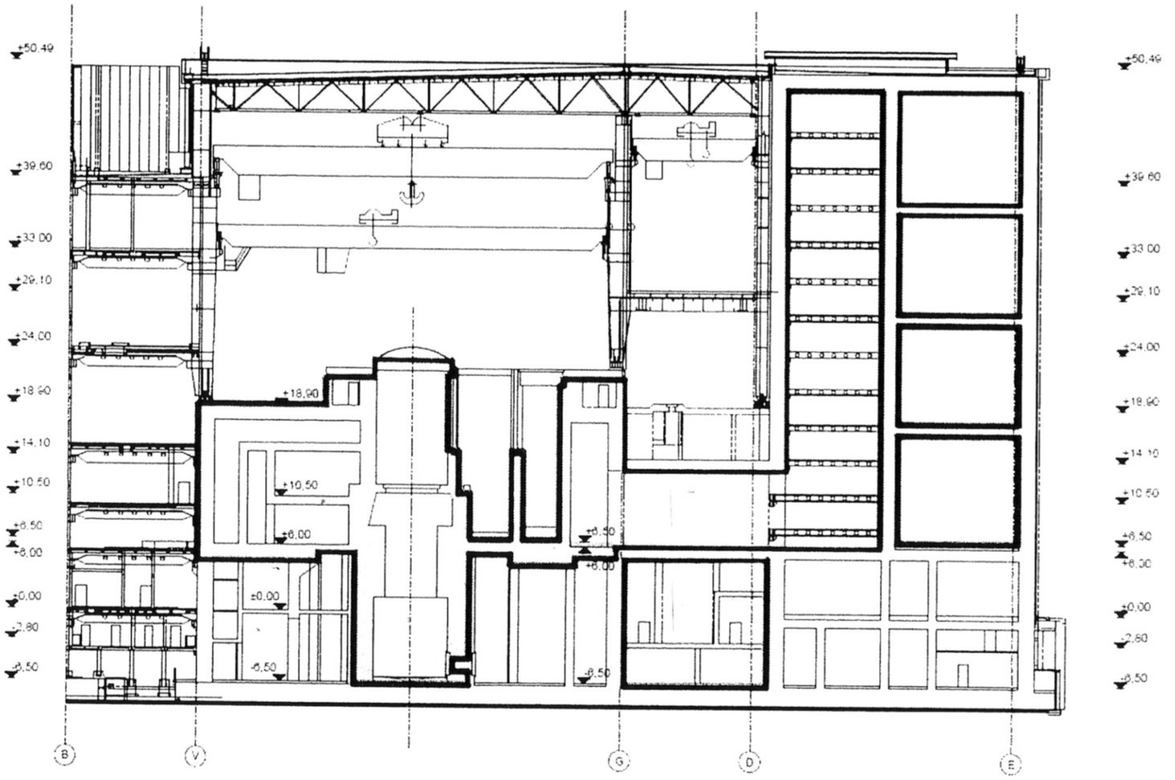
Cable groups of neutron flux measuring lines

No	Commodity group	Type group	Cable No.	Cable length m	Description of qualified state and their justification	Qualification
31	CG_JE-YCY	JE-YCY	48	6995	Ambient temperature up to a maximum of 34 °C. Do not have functions during emergency conditions. Artificial Ageing: Thermal operating simulation: maximum 40 °C certified for 38 years Document: LOCA-19/2009-3	Pass: 359/VNL Product Certificate
32	CG_KMPEV	KMPEV	24	3305	Ambient temperature up to a maximum of 34 °C. The cables are laid in numbered trails unaffected by emergency effects. Qualification is done according to those described in row 6.	Pass: V01-24761_C11-2010, 412/VNL Product Certificate
33	CG_KPETI	KPETI	144	6945	Ambient temperature up to a maximum of 27 °C. The cables are laid in numbered trails in cable spaces of the secondary circuit, they are unaffected by emergency effects. Artificial Ageing: Thermal operating simulation: maximum 55 °C certified for 50 years Document: Y-22012007-6	Pass: 350/VNL Product Certificate
34	CG_NHXHX	NHXHX Siepopyr	12	1655	Ambient temperature up to a maximum of 32 °C. The cables are laid in numbered trails in cable spaces, they are unaffected by emergency effects. Based on the description in row 17, their qualification is satisfactory.	Pass: 362//VNL Product Certificate
35	CG_NU-HXH	NU-HXH	288	11454	Ambient temperature maximum 49 °C. Emergency conditions: T emergency=108°C, T sbk=120°C Artificial Ageing: Thermal operating simulation: 110 °C/35 days/23 years Gamma operating simulation: 52.6 kGy, Gamma emergency: 33 kGy Tmax emergency: 160 °C Document: LOCA-19/2009-5	Pass: 400/VNL Product Certificate
36	CG_NYY	NYY	24	3585	The cables concerned, operate normally in ambient temperature up to a maximum of 34 °C. From the operating temperature aspect, their compliance until the end of the service life extension is shown in row 22 of this table.	Pass: 365/VNL Product Certificate

1.4 ANNEX 05.1.1-1: RPV DEGRADATION LOCATIONS



1.5 *ANNEX 07.1.1.1-1: LAYOUT OF THE REINFORCED CONCRETE CONTAINMENT*



Hermetic steel lining

The reactor building and the localization tower (on the right side) can be seen in this figure. The bold line is the hermetic liner which is the boundary of the hermetic compartments (containment).