Full Paper Transactions, SMiRT 28
Toronto, Canada, August 10-15, 2025

AGEING MANAGEMENT ANALYSIS OF VVER REACTOR INTERNALS FOR LONG-TERM OPERATION BEYOND 60 YEARS IN THE CZECH REPUBLIC

Vladislav Pištora[1], Petr Vlček[2], Miroslav Žamboch[3]

[1] Expert of Structural and Lifetime Assessment, UJV Rez, Inc., Rez, Czech Republic (vladislav.pistora@ujv.cz)
[2] Research and Development Specialist, UJV Rez, Inc., Rez, Czech Republic (petr.vlcek@ujv.cz)
[3] Structural and Lifetime Assessment Section Head, UJV Rez, Inc., Rez, Czech Republic (miroslav.zamboch@ujv.cz)

ABSTRACT

This paper describes the Ageing Management Program (AMP) applied to reactors operated in the Czech Republic, specifically to the reactor vessel internals (RVI). The AMP aims to demonstrate the remaining service life of the assessed components and equipment, ensuring the integrity and required functions of the components throughout the operation of the nuclear power plant (NPP). If necessary, it initiates the timely implementation of effective corrective measures to guarantee the expected long-term operation.

The status assessment is performed for selected degradation mechanisms expected in the reactor internals, with the requirement to understand the current status and predict its possible development. For this purpose, in-service inspection program and maintenance activities is compiled.

INTRODUCTION

Currently, to minimize costs and to ensure the stability of energy supply, efforts are being made to maximize the time of operation of existing nuclear power plants. The Ageing Management Program (AMP) serves to fulfil these goals. The AMP involves activities, analyses, and evaluations to assess the condition and aging of components. Its goal is to ensure the integrity and sufficient residual life of components throughout the operation of a nuclear power plant and to implement corrective actions if necessary.

The analysis consists of continuous periodic evaluation of the impact of degradation mechanisms and aging effects on the material of the assessed component. This is carried out depending on the mode of operation and operating conditions. Timely detection of excessive degradation mechanisms or aging effects is ensured by regular evaluations on an annual basis. The evaluation of degradation mechanisms and aging effects on the performance of the required function of the equipment is carried out using defined parameters and their limit values. This evaluation timely indicates an excessive trend of equipment damage. When degradation mechanisms and effects of aging are manifested, it is necessary to take into account the given knowledge and proceed to update the activities monitoring their development, or to take other corrective measures.

The AMP involves various activities such as regular inspections, testing, analyses, and assessments of the condition of components. These activities help identify and prevent potential problems caused by the aging of materials and equipment. The program also includes the implementation of corrective measures, if necessary, to ensure the integrity and functionality of the components throughout their operation.

AGEING MANAGEMENT PROGRAM FOR THE REACTORS IN THE CZECH REPUBLIC

The aim of the AMP is to ensure the safety and functionality of nuclear power plant components during operation and aging of the equipment, taking into account the required duration of operation. The AMP criteria are related to the conditions set for operation by the designer and manufacturer of the equipment.
The aging management program is based on the International Atomic Energy Agency (IAEA) methodology (IAEA (2018)) and is the same for all units in the Czech Republic, considering the technical differences between the reactor internals of VVER 440 and VVER 1000. The IAEA methodology for an Aging Management Program (AMP) includes several key elements that are crucial for the effective management of aging components in nuclear power plants:

- Identification of systems, structures and components for which AMPs are required,
- Identification of degradation mechanisms and effects of aging, their dependence on operating conditions, and determination of their manifestations,
- Establishment of methods for detecting aging effects before they lead to failures, including non-destructive testing, visual inspections, and monitoring systems,
- Implementation of measures to prevent or mitigate aging effects in line with maintenance and upgrade activities,
- Corrective actions - development of procedures to address identified aging effects, including repairs, replacements, or modifications,
- Ensuring adequate documentation of activities carried out to enable trending, monitoring of the development of manifestations, etc.,
- Application of technologies and adjustments to operating conditions based on operating experience and new knowledge.

These activities help ensure that aging effects are identified in a timely manner and effectively managed while ensuring the safety and reliability of power plant operation.

The requirements for the evaluation of VVER reactors are based on the manufacturer's project. The Dukovany and Temelin NPPs in the Czech Republic were designed by Czech engineers in collaboration with Soviet experts. The main designer was the company Skoda Praha, which specializes in the design and construction of power equipment. The manufacturer of the reactor pressure vessels (RPV) was Skoda JS. The reactor internals were supplied by other manufacturers.

AGEING MANAGEMENT PROGRAM FOR REACTOR INTERNALS

The reactors internals play a crucial role in ensuring safe and efficient NPP operation. Unlike the reactor pressure vessel, these parts do not provide a pressure boundary and do not form a physical barrier between the operating medium and the external environment. Nevertheless, very high safety requirements are placed on them, especially in the core area, where specific operating conditions differ from the remaining parts of the primary circuit.

The reactor internals perform several important functions, including:

- Directing coolant flow: It is essential for effective heat removal from the reactor core.
- Radiation and thermal shielding: RVI protect other parts of the reactor and personnel from harmful radiation.
- Defining the geometry of the reactor core: It is important for the proper functioning of the reactor.
- Resistance to degradation: RVI materials are designed to withstand radiation and thermal stress, ensuring long-term reliability.

The Ageing Management Program for RVI is based on the methodology provided by the IAEA, specifically IAEA (2018) and the International Generic Ageing Lessons Learned (IGALL) program.

Periodic status assessments are carried out in accordance with the AMP. The condition of the reactor internals is assessed annually through detailed inspections and assessments to identify any signs of wear, degradation, or damage. Techniques such as non-destructive testing, visual inspections, and monitoring systems are used to ensure a thorough assessment.

Some analyses, particularly those relating to long-term degradation mechanisms, are updated every five years to incorporate the latest data and technological advances into the ageing management strategy.

Regular condition assessments provide a comprehensive understanding of the current state of the reactor internals, identifying existing issues and evaluating the effectiveness of previous maintenance activities. The program includes predictive analyses to forecast the development of degradation over time, which is crucial for planning maintenance and replacement activities. Predictions currently extend up to 60 years of operation, with ongoing efforts to extend them up to 80 years.

Based on condition assessments and predictive analyses, the program provides recommendations to ensure the safe operation of the reactor internals for the required duration. These recommendations may include specific maintenance activities, upgrades, or modifications to address identified ageing effects.
By following these detailed steps, the AMP ensures that ageing effects are managed effectively, maintaining the safety and reliability of nuclear power plant operations.

Based on information from the design, operational experience, and information from abroad, the following degradation mechanisms and aging effects have been identified or are expected for the reactor internals:

1. Fatigue - localized structural damage leading to crack formation caused by cyclic loading. This is a common issue in reactor internals subjected to repeated stress cycles.

2. Vibration Fatigue - caused by vibrations due to pressure pulsations from the main circulation pumps. These vibrations induce stress cycles in the RVI material, leading to the formation of microscopic cracks that grow over time and eventually lead to failure.

3. Wear and Mechanical Damage - results from impacts, abrasion, or other physical interactions, leading to surface degradation and potentially to the formation of cracks. Damage occurs at the points of mutual contact between reactor internals components or with the RPV. Damage caused by handling of the reactor internals during their disassembly/assembly during fuel changes is also common.

4. Stress Corrosion Cracking (SCC) and Irradiation-Assisted Stress Corrosion Cracking (IASCC) - growth of cracks in a corrosive environment, enhanced by tensile stress. Normally, materials used for reactor internals are insensitive to crack development by SCC, but conditions may occur where a specific type of SCC is accelerated by neutron irradiation, making the materials more susceptible to stress cracking in the reactor environment.

5. Loss of Fracture Toughness - thermal aging and neutron irradiation can degrade the fracture toughness of reactor internals materials. This brittleness makes the material more susceptible to fast fracture, particularly in areas known as limited embrittlement areas (LEAs).

6. Radiation Swelling and Creep - radiation swelling involves an increase in volume and dimensional changes of materials due to the accumulation of crystallographic lattice defects induced by radiation. Creep is the slow, time-dependent deformation of materials under constant stress and high temperature. Both phenomena can significantly change the mechanical properties and dimensions of reactor internals materials over time., The conditions in reactor interiors (neutron dose, temperature) are conducive to their development.

The computational assessment of the integrity and lifetime of RVI is currently conducted in the Czech Republic in accordance with the NTD (2023) documentation. Previously, the RVI assessment was carried out using preliminary versions of VERLIFE document (see its current version: Brumovsky M., Martin O. (2024)).

The implemented procedure is used for the assessment of RVI for VVER-type reactors. The scope of applicability allows for the assessment of the lifetime of RVI according to the following criteria:

- The initiation and growth of cracks through fatigue mechanisms and/or stress corrosion cracking in the presence of radiation (IASCC).
- The formation of limit embrittlement areas (LEA) due to neutron radiation.
- Stable crack growth, unstable crack propagation, and loss of component load-bearing capacity.
- Changes in RVI dimensions due to radiation swelling.

For the assessment of the RVI status and development of degradation mechanisms, including their prediction, it is necessary to know the operating conditions and history of operation as accurately as possible. This applies in particular to the integrity of the component.

Therefore, monitoring and determination of neutron doses on the RVI are carried out, including prediction for future operation. The determined trend curves indicate the dependence of the dose on time of operation and are used in the relationships for calculating material characteristics based on the operating time for selected parts of RVI. Based on the determined fluences (doses), the calculation of radiation heating for individual components of the RVI is carried out, thus enabling the prediction of operating parameters.

Another important input for the implementation of most computational assessments for various aging effects such as crack growth, crack initiation, loss of load capacity, and change in geometric dimensions is information on the mechanical properties of the material corresponding to the operating time.

All these degradation mechanisms and aging effects are analysed using available analytical methods, and affected areas are identified based on their potential development. In areas with potential degradation, regular inspections and advanced monitoring techniques are implemented to detect any signs of degradation mechanisms.

REACTOR INTERNALS OF VVER 440 NPP DUKOVANY

In the VVER 400 type reactor, the reactor internals consist of core barrel, core barrel bottom, core basket and protective tube unit. The individual parts are shown in Figure 1.

The following degradation mechanisms are evaluated for RVI of NNP Dukovany:

1. Low cycle fatigue is evaluated for all RVI components in selected construction areas. The influence of the environment and irradiation is considered. The lifetime of 60 years of operation is guaranteed for VVER 440 RVI in their current state. The process of lifetime assessment beyond 60 years is ongoing. The remedial action is necessary to increase the knowledge of the fatigue behaviour of the material and possibly reduce unnecessary conservatism in the analysis.

2. IASCC - A computational assessment of possible crack nucleation by the irradiation-assisted stress corrosion cracking mechanism was performed for 40 and 60 years of operation. A simplified methodology according to NTD (2023) was used for the three most vulnerable RVI components, such as the core barrel, core basket, and bottom plate of the protective tube unit. The assessment was performed only for the "full power state" mode, which is significant in terms of the internal heat generated by gamma radiation and the duration of the mode.

For the monitored RVI, it was found that the IASCC threshold curve is not exceeded and therefore the full power state does not contribute to the risk of crack nucleation by the IASCC mechanism.

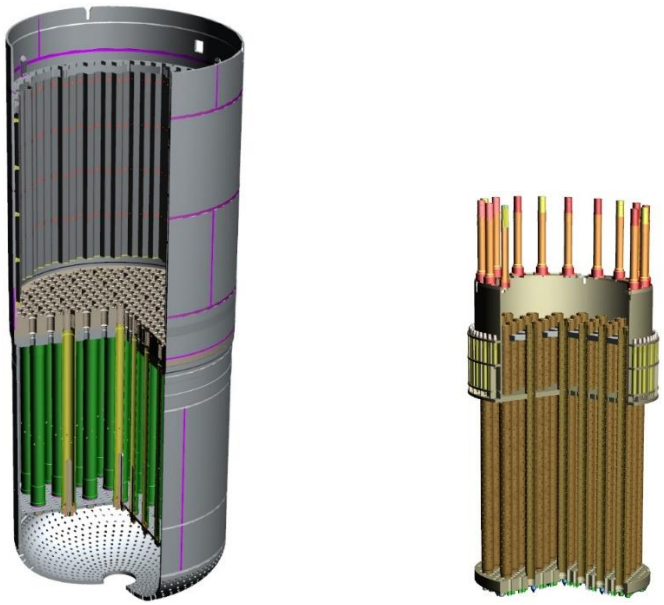

Figure 1. Reactor internals of the VVER 440 type reactor (barrel, core barrel bottom, core basket – left, protective tube unit – right)

Other locations potentially sensitive to IASCC (based on information from abroad) are the core basket bolts. The evolution of stress at the evaluated points of the bolts and washers during the NPP operation (in dependency on neutron dose increasing during the NPP operation and expressed in terms of displacement per atom – dpa) is shown in Figure 2. It is evident from this figure that at the beginning of operation the absorbed dose is < 3 dpa and thus one of the conditions for the formation of IASCC is not met. After 2nd campaign, the stresses decrease significantly by stress relaxation due to irradiation creep, and again the IASCC threshold is not met.

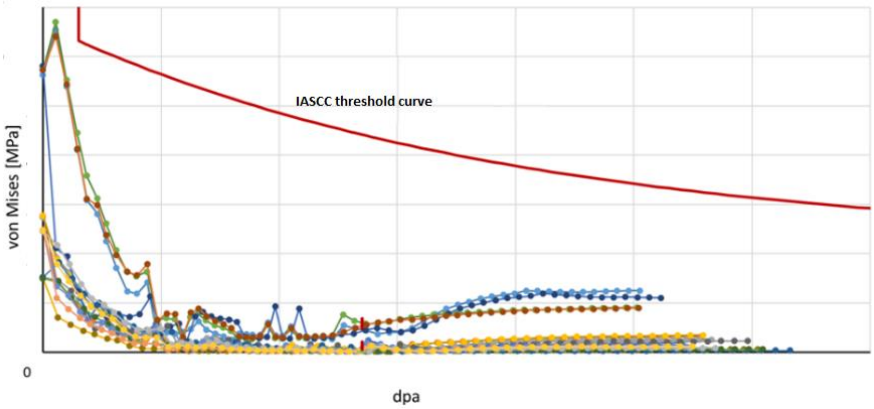

Figure 2. Example of evaluation of selected points for IASCC initiation for RVI of typical VVER 440

3. The vibrational state of RVI is monitored for VVER 440 by a diagnostic system with displacement sensors. The actual vibrational state is compared with existing reference vibrational masks provided by the designer, based on previous operation. For LTO 60+, the diagnostic systems should be modernized with new sensors applied. An assessment of the susceptibility of the RVI to potential damage due to high cycle fatigue is performed for safety-relevant nodes of the RVI components. Inputs are taken from the dynamic model of RVI and pressure fluctuation measurements.

4. Wear is assessed in selected structural joints where two reactor components come into contact, such as the tongue-and-groove joints. Assessment of wear and potential mechanical damage due to aging caused by normal operation is performed using two primary methods:

- Vibration diagnostics: This method involves monitoring the vibrational state of components to detect any anomalies or changes that may indicate wear or mechanical damage.
- In-service inspections (ISI) - television visual inspection method: This method involves visual inspection of components using television cameras or other visual inspection tools (see example in Figure 3).

The combination of these two methods provides a comprehensive assessment of wear and potential mechanical damage, ensuring that any problems are detected early and promptly addressed to maintain the reliability and safety of the structural joints.

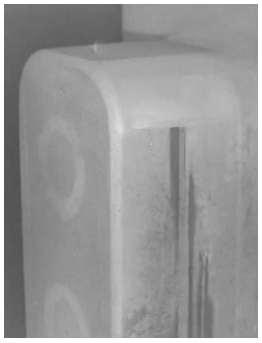

Figure 3. Mechanical damage on the console tongue

5. Damage of irradiated materials by the degradation mechanisms of radiation swelling and creep predominantly occurs at temperatures around 400°C and higher. The degree of damage depends on the chemical composition, temperature, and especially on the neutron dose. Results of experiments on the RVI material from the decommissioned VVER-440 nuclear power plant after 45 years of operation showed negligible radiation swelling at irradiation temperatures of 280 - 315°C and dose below 43 dpa. For PWR RVI, where the irradiation temperature is below 380-390°C, an increase of radiation swelling greater than 0.07%/dpa is not expected, which corresponds to less than 5% radiation swelling strain over 60 years of operation.

Nevertheless, computational assessments of swelling development are also carried out for RVI of Dukovany NPP. The calculations were performed in detail based on the results of neutron dose calculations, which were established for all completed and predicted campaigns up to a period of 60 campaigns of operation. The assessment confirmed minimal swelling development in the VVER 440 RVI components.

REACTOR INTERNALS OF VVER 1000 NPP TEMELIN

In the VVER 1000 type reactor, the reactor internals consist of core barrel, core baffle and protective tube unit. The individual parts are shown in Figure 4.

Figure 4. Reactor internals of the VVER 1000 type reactor (core barrel - left, core baffle – center, protective tube unit – right).

The RVI assessment for the Temelin NPP is carried out in a very similar manner to that for the Dukovany NPP. The results are also similar, so there is no need to comment on them again. Due to the higher power and slightly different dimensions of the reactor pressure vessel, some components have a different design. This affected the design of the reactor core and core baffle (which substantially differs from core basket of VVER 440) and also had an impact of the degradation mechanism of swelling and creep. Due to higher neutron dose and higher core baffle thickness (which leads to higher temperature inside its wall), there is a demonstrable development of this degradation mechanism.

The initial assessment of radiation swelling is carried out using simulations that model the manifestations of this degradation mechanism. The computational evaluation is based on input data from actual operation (core configuration), and the development of swelling is predicted up to 80 years of operation.
Swelling causes macroscopic changes of the core baffle geometry, resulting in deformation of the original shape and radial/axial displacements. Differences in the burnout of the fuel assemblies and their positions around the core periphery cause a non-uniform distribution of the temperature field. The non-uniformity of swelling and creep volumetric strains leads to radial displacements of core baffle surfaces.

The radial displacements of the most affected core baffle areas are highlighted in Figure 5. Parts of the core baffle move toward the centre of the reactor core, while other parts move toward the core barrel. In addition to radial deformation, there is also a small horizontal deformation of the core baffle rings.

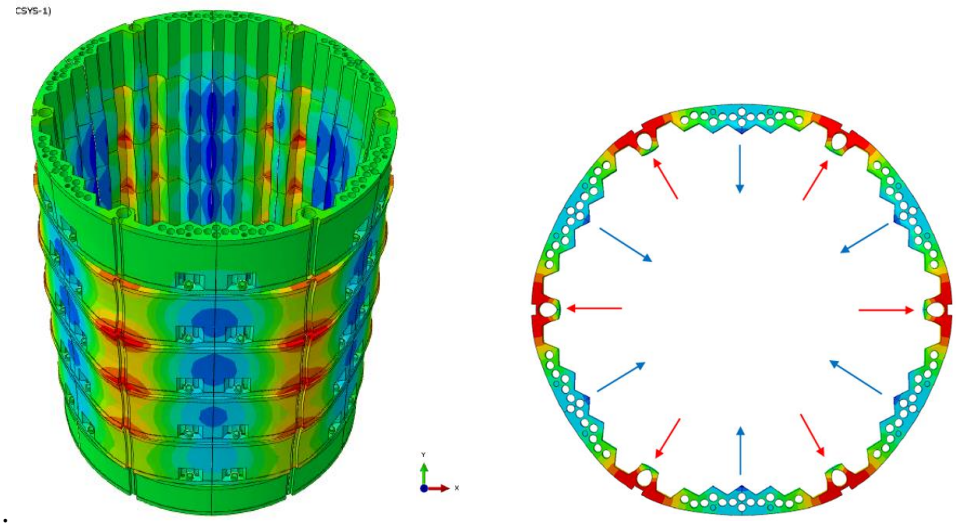

Figure 5. Core baffle deformation from radiation swelling (cold state)

The method for measuring corresponding changes in core baffle geometry was selected, and a measuring device was designed and fabricated. This measurement provides information about the real physical condition of the components concerning the swelling development.

NON-DESTRUCTIVE TESTING OF REACTOR INTERNALS

Non-Destructive Testing (NDT) plays a crucial role in the maintenance and safety assessment of nuclear reactor internals. It involves various techniques to inspect and evaluate the status of reactor components without causing any damage. The primary objectives of NDT are to identify degradation mechanisms (DM) and ensure the long-term operation of reactor internals.

To identify the manifestations of degradation mechanisms, the following inspection methods are prescribed within the in-service inspection program for reactor internals in the Czech Republic:

Visual Inspection is performed exclusively using remote methods, such as cameras or other imaging equipment to examine areas that are difficult to access. In terms of applicability of the method, it is the most widely used for RVI inspection and is used to identify all types of surface defects, including cracks protruding to the surface, identification of foreign objects, etc. The disadvantage is the subjective interpretation of the results.

The second very common inspection method is dimensional inspection, which is applied mainly during the assembly and disassembly of RVI. It ensures that all components are in their position. The measurement of changes in geometry caused by the swelling degradation mechanism is specific.

Other methods, such as ultrasonic testing and eddy currents, are used in specific cases. The application of these methods to RVI components requires specific equipment due to testing conditions (high doses, work under water). Due to the importance of inspections, they are in most cases performed using a qualified inspection procedure to demonstrate the ability of the NDT system to identify the damage and, if necessary, to determine the dimensions of the defect with the required accuracy.
All inspections are performed periodically according to the potential for defect development, ranging from each shutdown to a period of up to 8 years.

Sensors monitoring the condition of the reactor during operation are also part of the status assessment. This technique provides real-time data on parameters such as vibration, temperature, and pressure, which helps to identify potential problems in a timely manner.

Using these ISI techniques, the integrity and safety of reactor internals can be effectively monitored and maintained, ensuring long-term reliable operation of nuclear reactors.

An example of dimensions measurement application is the determination of the core baffle dimensions. The aim of the measurement is to verify the development of swelling and validate the computational models. It was decided to measure the distance between selected opposite inner surfaces of the core baffle. The same measurement was made during the production inspections. This makes it possible to compare the pre-operational and current dimensions of the core baffle.

The measurement of the distance between selected inner surfaces of the core baffle (see Fig. 6) is performed at different selected height levels. Measurements are taken in three angular positions, with rotations of 60° each time. Model and photo of the measuring device is seen in Fig. 7.

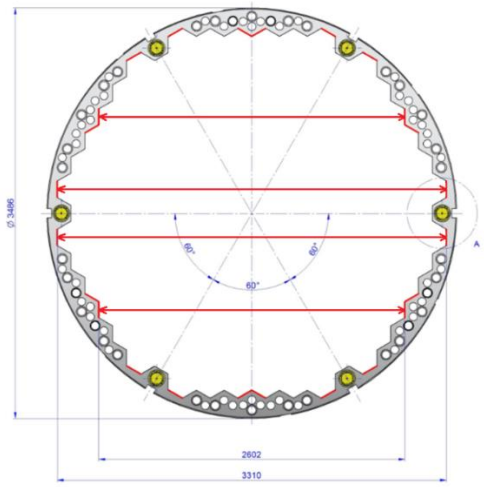

Figure 6. Positions, where the measurements of distance of opposite inner surfaces were carried out

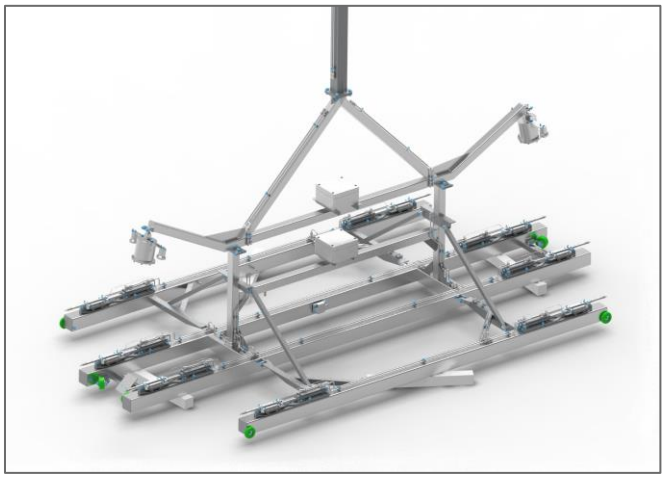

Figure 7. Model of measuring device

Due to the requirement to know the actual condition of the reactor internals for operation of 60 years or more, it is necessary to carry out inspections of these components. Information on the current state of the core baffle will allow for assessing the condition of this component and predicting the impact of the degradation mechanism on further operation in a timely manner.

CONCLUSION

The Aging Management Program is a comprehensive set of activities and procedures aimed at identifying and controlling the aging mechanisms of components and systems important for the safety of nuclear power plants. In the Czech Republic, the AMP program is implemented for VVER 400 and 1000 NPPs. Degradation mechanisms that can affect safety-critical components are identified and qualified based on design specifications, operating conditions, operational data, and knowledge from abroad.

Using computational and analytical tools, the locations of RVI with the greatest potential for the development of degradation mechanisms are analysed and evaluated. At the same time, operational inspections and maintenance activities are carried out to examine the real condition of RVI and identify possible damage development. A suitable example is measuring the geometry of the core baffle.

Based on these assessments and findings, measures to prevent or mitigate the negative effects of aging are implemented. The Aging Management Program is crucial for the long-term safe operation of nuclear power plants and ensures that all components remain in optimal condition throughout their life.

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