**AMP 162 PWR REACTOR PRESSURE VESSEL (Version 2020)**

**Programme Description**

This ageing management programme (AMP) is a component-specific programme for the reactor pressure vessel (RPV) components of PWRs, including WWERs, that covers degradation mechanisms of the RPV (reactor vessel, lower and upper heads) and the activities necessary to manage the ageing mechanisms. As such, this AMP refers to other degradation-specific AMPs that deal with particular degradation mechanisms and ageing effects (AMP 101, AMP 110, AMP 111, AMP 118, AMP 152 and AMP 160).The reactor vessel is cylindrical with a hemispherical lower head (bottom dome) and a flanged and gasketed upper head. The reactor vessels have been fabricated by two methods: a) rolled plates welded with longitudinal and circumferential welds (PWR) and b) forged cylindrical shells with circumferential welds (PWR and WWER). However modern PWR reactor vessels do not have longitudinal welds in their constructions. Shell sections are made from individual ring forgings. Most new PWR vessels are fabricated from sufficiently large forgings that the mid-section of the core is covered by a single forging and the circumferential welds are displaced to regions of lower neutron flux. Also modern WWERs reactors do not have welds in the upper head and bottom dome (e.g. WWER-1000, WWER-1200) except for the circumferential welds which connect the upper head to the flange and bottom dome to the lower shell. The upper head is joined to the vessel by a flanged and bolted joint

The material of the vessel and heads is of low-alloy carbon steel. To minimize corrosion, the inside surfaces in contact with the primary coolant are clad with austenitic stainless steel. The cladding is applied in one or two layers by multiple-wire, single-wire, strip-cladding, or resistance welding processes. There are some WWER-440 RPVs (e.g. Russian Novovoronezh 4 NPP/V-179 and Kola 2 NPP/V-230) which do not have austenitic stainless steel cladding.

The programme includes activities for inspecting, detecting, preventive, monitoring, mitigating and evaluating of the ageing degradation effects of the RPV components of the PWR and WWER reactors [1, 2].

### Evaluation and Technical Basis

1. ***Scope of the ageing management programme based on understanding ageing:***

The scope of the programme includes the reactor vessel, flange joint, upper and bottom heads, as well as RPV control rod drive mechanism (CRDM) and bottom-mounted instrumentation (BMI) nozzles. The scope of the programme does not include closure bolting components of the reactor vessel head (studs, washers, bushings, nuts and flange threads) because they are adequately managed in accordance with AMP104. The following ageing degradation mechanisms are considered in this AMP:

* Neutron irradiation embrittlement;
* Thermal ageing;
* Fatigue (low-cycle and environmentally assisted fatigue);
* Stress corrosion cracking (primary water stress corrosion cracking);
* General corrosion, crevice corrosion and pitting;
* Boric acid corrosion;
* Wear.

Neutron irradiation embrittlement

The materials of the reactor vessel (base metal, welds, corrosion-resistant cladding) are exposed to the radiation action of a fast neutron flux with energy greater than 0.5 MeV. The overall effect of fast neutron exposure is that ferritic steels experience an increase in hardness and tensile properties and a decrease in ductility and toughness, under certain conditions of radiation. The most critical area for irradiation embrittlement is the beltline region (the area opposite to the reactor core).

Thermal ageing

Thermal ageing is a temperature, material state (microstructure) and time dependent degradation mechanism that decreases material toughness. In the process of operation, the reactor vessel (base metal and welds) is exposed to high temperatures (260 – 325 °С), which may lead to thermal ageing and change the level of the mechanical properties and service characteristics of the material. The critical locations for thermal ageing are vessel shells (intermediate shell, lower shell including beltline weld, upper shell) and nozzles (e.g., inlet, outlet and safety injection). However, it should be noted that in an operating plant, thermal aging would be expected to be manifested earlier in life in the higher temperature operating pressurizer than in the reactor pressure vessel itself.

Low cycle fatigue

Low cycle fatigue is caused by cyclic loading of system, structures and components during operation. The critical components are vessel shells, flange, head nozzles, nozzle safe ends, penetrations, thermal sleeve and welds.

Primary water stress corrosion cracking

Primary water stress corrosion cracking (PWSCC) is defined as the intergranular cracking of nickel-base alloys that requires the presence of high applied or residual stress, susceptible microstructures (few intergranular carbides) a primary water environment and high temperatures. The critical components for cracking due to PWSCC are dissimilar metal welds of the head CRDM penetrations, bottom-mounted instrumentation (BMI) nozzles (for PWR), inlet (cold leg), outlet (hot leg) and safety injection nozzle safe ends. AMP111 and AMP163 are focused on managing the effects of cracking due to PWSCC of all susceptible nickel alloy-based components of the reactor coolant pressure boundary, including nickel-alloy butt welds (600/82/182).

General corrosion, crevice corrosion and pitting

This type of corrosion can be observed on the inside surface of the reactor vessel when protective austenitic stainless steel cladding has not been applied. All unalloyed or low alloyed ferritic steels are subject to the formation of a magnetite protective layer as a consequence of the reaction between the water and the steel at operating temperature. Nevertheless, large scale surface corrosion and pitting was observed in most of these vessels (in the core region and in the nozzle to safe-end zone). This corrosion was attributed to oxygen pick-up by the PWR water during shutdown periods with the oxygen remaining in the primary system for a period after startup, rather than the corrosion occurring in the presence of properly controlled water chemistry [2].

Boric acid corrosion

The boric acid corrosion is a potential degradation mechanism for carbon and low alloyed steels when in contact with external surface: RPV upper head and bottom dome (including nozzles). This could result from CRDM penetrations and bottom instrument tubes leakage, as well as leak from the flange seal.

Wear

Degradation due to wear may occur during maintenance operations concerned with opening and closing of the RPV head. The sealing faces of the RPV flanges are subject to ageing degradation and require to be cleaned, and potentially repaired as part of normal maintenance. The degradation can be detected by visual inspection long before the effects of wear begin to compromise the RPV structural integrity.

1. ***Preventive actions to minimize and control ageing degradation:***

The programme identifies the preventive actions defined as those that are necessary to prevent or minimize initiation of degradation during normal operation.

The operator's practices have an influence on the operating parameters of the reactor installation and their actions have a significant role for implementing the programme for decreasing the degradation of the RPV metal through management of the allowable limits. For example, preventive actions during unit startup and shutdown:

* Observance of temperature and pressure restrictions during reactor heating up and shutdown cooling;
* Prevention of thermo-hydraulic transients and conditions leading to pressurized thermal shock.

Predictive maintenance and repair (M&R) via developing of the M&R long-term schedule (periodicity and scope of activities).

Low cycle fatigue, PWSCC and reducing of boric acid corrosion preventive actions are addressed in relevant programmes: AMP 101, AMP 111 and AMP 110.

The preventive actions to eliminate potential adverse effects of water chemistry on the ageing mechanisms (corrosion) and to minimize contamination of components (surfaces) from carbon and low alloyed steels by primary water are carried out during normal operation via monitoring and management of water chemistry conditions in accordance with the plant programme. The programme description, evaluation and technical basis for monitoring and maintaining of reactor coolant chemistry are addressed in AMP 103.

This programme may also rely on minimization of neutron irradiation of RPV material using fuel management approaches such as low leakage zone configurations.

1. ***Detection of ageing effects:***

For each ageing effect of RPV, the state parameters to be controlled are determined, as well as the technique for performing controls to ensure the ageing management of the elements under consideration.

The in-service inspections are intended to detect cracking due to PWSCC, crack growth due to cyclic loading, loss of material due to boric acid corrosion, general corrosion, crevice corrosion or pitting and wear. The following non-destructive examination (NDE) methods may be used according to the national regulations (e.g., [3-6]) and recommendations addressed in AMP 102: visual testing (VT), dye penetrant testing (PT) or magnetic particle testing (MT), eddy current (ECT) and ultrasonic testing (UT). In addition to RPV internal remote UT, some IAEA Member States countries operating WWER-1000 also carry out RPV external examination by remote UT of the under supporting shell welds and by manual examinations of the over supporting shell welds, such as UT, VT and PT/MT. The frequency and scope of NDE shall be regulated in accordance with national NDE programmes/instructions. The inspections and monitoring are performed by qualified personnel using qualified techniques in accordance with approved licensee procedures. The reliability of flaw detection and sizing are confirmed by the UT system qualification of the RPV components in accordance with [4, 6-9].

The management of thermal ageing and neutron irradiation embrittlement is carried out by RPV surveillance programmes. The objective of the reactor vessel material surveillance programme is to provide sufficient surveillance specimens and dosimetry data to:

* Monitor irradiation and thermal embrittlement until the end of the operation period;
* Determine the need for operating restrictions (e.g., operating pressure and temperature limits, and the neutron fluence limit for continued operation).

The programme description, evaluation and technical basis of the RPV surveillance programme are addressed in AMP 118, AMP 152 and AMP 160.

The cumulative effect of fatigue is addressed by AMP 101.

1. ***Monitoring and trending of ageing effects:***

The methods for monitoring, recording, evaluating, and trending the data that result from the programme’s inspections shall provide for identification of adverse ageing trends such that corrective action can be performed as necessary in a timely manner.

Fracture toughness reduction monitoring

The reduction of fracture toughness of reactor vessel beltline materials occurs due to both neutron irradiation embrittlement and thermal embrittlement and the long term operating conditions (cold leg operating temperature and neutron fluence) that could affect the reactor vessel embrittlement. The programme description of mechanical properties monitoring by accumulated neutron fluence and surveillance specimens programme are addressed in AMP 118, AMP 152 and AMP 160.

ISI results from monitoring and trending

To facilitate monitoring and trending of RPV metal conditions the data of inspection results are collected, compared and assessed to make predictions for the future. For example, a comparison of the current UT results with previous ones is performed in order to determine the indications growth rate.

Actual number of cycles monitoring

Monitoring and trending the RPV operational history, i.e. the sequence of operational transients characterised by pressure and temperature time courses, can enable the periodical fatigue evaluation in critical locations of the RPV, as discussed in AMP 101.

1. ***Mitigating ageing effects:***

The activities referred to section 2 are implemented to mitigate degradation of ageing effects.

Neutron irradiation embrittlement can be mitigated by flux reduction (fuel management implementing a low neutron leakage core or shielding elements) or by thermal annealing of the reactor vessel.

1. ***Acceptance criteria:***

NDEs acceptance criteria are provided from applicable national regulations, codes, standards and guidelines [3-6].

Strength calculations (cyclical crack growth due to fatigue mechanism, resistance to brittle fracture for postulated or fixed discontinuities) are applied to estimate of critical size of the crack-like defect (e.g., [10 - 12]).

Acceptance criteria for mechanical properties are their compliance (conformity) to the RPV and surveillance specimen manufacturer data or Technical Specifications (for conservative assessment) for a given grade of material (e.g., Russian regulations [13, 14]).

Mechanical properties prediction methods of RPV irradiated welded joints are applied as described in AMP152.

The approaches to the RPV embrittlement acceptance criteria are addressed in AMP 118 and AMP 152 (for WWER).

The approaches to the neutron fluence values acceptance criteria are addressed in AMP 160. The activities for verification and validation of the neutron fluence at the RPV are presented in AMP 152.

The estimated number of loading cycles is defined in the list of operation modes of the reactor plant taking into account extended lifetime period.

Acceptance criteria of the fatigue cumulative usage factor "CUF", without and by taking into account the environment influence “CUFen”, are addressed in AMP 101.

1. ***Corrective actions:***

Corrective measures shall be carried out in accordance with the instructions, design rules, codes and standards applicable at NPP.

In order to satisfy the safety requirements, further evaluation to demonstrate fitness-for-service of the component until the end of the next periodic inspection interval may be required. Examination results and flaws that exceed the acceptance criteria given in the governing requirements or guidance documents may require analytical evaluation (e.g., new strength calculations [11, 12, 15]) for continued service until the next inspection, as well as supplementary examinations to further characterize the detected condition.

When suitable in some cases, changes in operational regimes could be applied.

1. ***Operating experience feedback and feedback of research and development results:***

This AMP addresses the industry-wide generic experience. Relevant plant-specific operating experience is considered in the development of the plant AMP to ensure the AMP is adequate for the plant. The plant implements a feedback process to periodically evaluate the plant and operating experience and research and development (R&D) results [16, 17], and, as necessary, either modifies the plant AMP or takes additional actions (e.g. develop a new plant-specific AMP) to ensure the continued effectiveness of the ageing management.

Appropriate sources of external operating experience are WANO Operating Experience Programme, IAEA IGALL Programme, etc. Effective experience exchange is an important element for implementing continuous improvement in this programme and in defining adequate corrective actions.

The operating experience in recovery of the RPV welds mechanical properties by thermal annealing. The pilot recovery annealing of the WWER-1000 reactor vessel core welds and base metal of unit 1 of the Balakovo plant was carried out in 2018. Previously, similar technology was tried and tested at WWER-440 reactor vessels only for core welds.

Experiencein the area of thermal ageing and irradiation embrittlement research is reviewed in AMP 152.

Incidents of PWR RPV penetrations and nozzles (CRDM vessel head penetration and BMI penetration nozzle) ageing degradation are identified in IAEA [Nuclear Energy Series](https://www.iaea.org/publications/search/type/nuclear-energy-series) NP-T-3.13 [18]. Operating experience with accelerated wear of CRDM thermal sleeves is described in EPRI Materials Reliability Program MRP-227 (Revision 1-A) [19].

Recent operating experience related to the control rod drive mechanism (CRDM) penetration thermal sleeve [20] is inconsistent with previous wear phenomenon [21] due to the failure location and fracture surface. This new type of ageing phenomenon indicates that cracks and eventual failure of the CRDM thermal sleeve can occur at multiple locations; thus, inspection guidance and recommendations were updated to address this recent experience [22-23].

In some WWER-440 plants geometrical instability (buckling) of the sleeve was observed (see also WANO report[[1]](#footnote-1)) which was not an integrity problem but might cause serious safety issue, i.e. by getting stuck the control rod. The reason of this phenomenon was that low-cycle fatigue during heating up and cooling down due to the different thermal physical properties of the materials took place in the upper welded joint between the penetration and the sleeve. As a result of the fatigue process, primary water penetrates through the microcracks into the gap between the tube and the sleeve. In the course of heating up the microcracks close and the medium’s thermal expansion generates stresses in the sleeve which exceed the yield strength of the sleeve’s material and lead to buckling. The root cause of the degradation was a design deficiency (extreme high CUF values were calculated in the vicinity of the weld). Since the weld position and geometry do not allow a proper NDE, a specific procedure had to be introduced. With UT it is possible to detect the water in the gap, and thus to identify the increased risk for deformation [24].

In the case of WWER-1000 reactor heads failure were observed in the lower welded joint between the penetration and the sleeve. Here a corrosion process in the penetration tube material beneath the sleeve was also detected which led to massive repair works. The root cause of the degradation was that inappropriate repair works of the welded joints were performed in the factory during manufacturing the reactor heads [25].

Experience feedback was done in performing of the RPV welds UT qualification. In 1998, IAEA published a document entitled “Methodology for Qualification of ISI System for WWER NPP” [7] as a suitable framework for developing credible evidence that ISI systems are capable to reliably detect and sizе flaws to meet ISI objective. Based on recommendations of the regional project RER/4/020 [26] (1998÷2003), a Pilot study to assist member states with practical advice was organized. The key feature of the Pilot study was based upon a real component - WWER-1000 RPV core region welds. A project team, led by IAEA with EPRI consultant, was set up with members representing Plant, Regulatory Body, Main Designer, Qualification Body and ISI Vendor.

1. ***Quality management:***

The AMP is carried out in agreement with site QA procedures, review and approval processes, and administrative controls, which are implemented in accordance with the different national or international regulatory requirements and standards, e.g. [27-30].

The inspections and monitoring are performed by qualified personnel using qualified techniques in accordance with approved licensee procedures. For example, the ultrasonic system, (devices, tools, techniques and experts) for RPV welding joints inspections, is qualified in accordance with [4, 7-9]. The personal performing these inspections are certified by the international or national legal organization, e.g. [31, 32].

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