## AMP 111 PWR Cracking of Nickel Alloy Reactor Coolant Pressure Boundary Components (Version 2021)

### Programme Description

This programme addresses the issue of cracking of nickel-alloy components and consequential loss of material due to boric acid-induced corrosion in susceptible, safety-related components in the vicinity of nickel-alloy reactor coolant pressure boundary components.

Regulatory authorities imposed long-term inspection requirements for the PWR vessel, steam generator, pressurizer components and piping if they contain the primary water stress corrosion cracking (PWSCC) susceptible materials designated alloys 600/82/182 [1, 2]. New requirements were also imposed for the long-term inspection of reactor pressure vessel upper heads with Ni-alloy penetrations and bottom mounted instrumentation tubes (BMI).

For example, the U.S. Nuclear Regulatory Commission (US-NRC) addresses inspection requirements for all Ni-alloy components in the reactor coolant pressure boundary of pressurized water reactors in Sections 50.55a(g)(6)(ii)(D), (E) and (F) of Title 10 of the US Code of Federal Regulations [3], which mandates the use of ASME Code Cases N-729-4, N-722-1 and N-770-2 with certain conditions.

The impact of boric acid leakage from non-nickel alloy reactor coolant pressure boundary components is addressed in AMP 110.

### Evaluation and Technical Basis

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

The programme is 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 welds. The programme also manages the consequential loss of material due to boric acid corrosion in susceptible components in the vicinity of nickel-alloy components. These components could include, but are not limited to, the reactor vessel components (reactor pressure vessel upper head, nozzle-to-pipe connections, instrument penetrations), steam generator components (nozzle-to-pipe connections, instrument connections, drain tube penetrations and divider plates), pressurizer components (nozzle-to-pipe connections, instrument connections, and heater penetrations), and reactor coolant system piping (instrument connections and full penetration welds).

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

This programme is a condition monitoring programme. Maintaining high water purity reduces susceptibility to PWSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the water chemistry programme. The programme description and the evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in AMP 103. An identification of the most sensitive components can be established in order to define a preventive non-destructive examination campaign (material index to evaluate the crack initiation susceptibility based on fabrication, chemical composition, residual and operating stresses, temperature).

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

This is a condition monitoring programme that monitors cracking/PWSCC for nickel-alloy components and loss of material by boric acid corrosion for potentially affected steel components. Reactor coolant pressure boundary cracking and leakage are monitored by the in-service inspection programme in accordance with applicable regulatory requirements and industry guidelines. Boric acid deposits, borated water leakage, or the presence of moisture that could lead to the identification of cracking or loss of material can be monitored through visual examination. The programme detects the effect of ageing by various methods, including non-destructive examination techniques. Reactor coolant pressure boundary leakage can be monitored through the use of radiation air monitoring and other general area radiation monitoring, and technical specifications for reactor coolant pressure boundary leakage. Evidence of reactor coolant leakage may manifest itself by the presence of boric acid residues.

The specific types of non-destructive examinations are dependent on the component’s susceptibility to PWSCC and its accessibility to inspection. Inspection methods, schedules, and frequencies for the susceptible components are implemented in accordance with applicable regulatory requirements and industry guidelines.

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

Reactor coolant pressure boundary leakage is calculated and trended on a routine basis in accordance with technical specification to detect changes in the leakage rates. An example of a boric acid corrosion management guidance is in [4].

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

Several methods are available to mitigate the risk of PWSCC of nickel alloy base metal and welds (weld overlay, stress improvement process, surface treatment, replacement of components with more PWSCC-resistant materials, water chemistry improvement by zinc injection or hydrogen adjustment).

1. ***Acceptance criteria:***

Acceptance criteria for all indications of cracking and loss of material due to boric acid-induced corrosion are defined in applicable regulatory requirements and industry guidelines. EPRI has developed specific guidance for bottom-mounted nozzles in [5]. Also, the initial technical bases for code case N-770-2 can be found in [6].

1. ***Corrective actions:***

Relevant flaw indications of susceptible components within the scope of this programme found to be unacceptable for further services are corrected through implementation of appropriate repair or replacement. In addition, detection of leakage or evidence of cracking in susceptible components within the scope of this programme require scope expansion of current inspection and increased inspection frequencies of some components, as required by applicable regulatory requirements and industry guidelines.

Repair and replacement procedures and activities comply with the applicable Codes (for example ASME Section XI in the U.S. [7] or RSE-M Code in France [8]).

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 plant and industry-wide operating experience and research and development (R&D) results, 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.

This programme addresses reviews of related operating experience, including plant-specific information, generic industry findings, and international data. Within the current regulatory requirements, as necessary, the operator maintains a record of operating experience through the required update of the facility’s in-service inspection programme in accordance with the applicable regulatory requirements and industry guidelines.

Cracking of Alloy 600 has occurred in PWRs [9-11]. Furthermore, ingress of demineralizer resins also has occurred in operating plants [12] which can exacerbate crack initiation and growth. The Water Chemistry programme, AMP 103, manages the effects of such excursions through monitoring and control of primary water chemistry. PWSCC also is occurring in the vessel head penetration nozzles ([13-16] for US PWRs). To date, however, there are no known instances of cracking of Alloy 690 penetrations on PWR reactor vessel heads.

In France, empirical initiation model and crack-growth rate from R&D programme are used to evaluate the risk of initiation and the risk of leakage when cracks are detected. In case of cracked components, studies are performed in hot cells of removed components to validate the pertinence of non-destructive examination.

To support analyses of cracking in Alloy 600 penetration containing reactor vessel heads EPRI has compiled up-to-date approaches for the calculation of crack growth rates in Alloy 600 and its weld metals Alloys 82 and 182. This updated approach includes quantitative factors for the effects of low stresses (the elimination of a previous threshold effect) and water chemistry effects. It also provides a probabilistic basis for accounting for heat-to-heat variations of properties. [17]. Corresponding data, also published by EPRI, indicate that Alloy 690 and its weld Alloys 52 and 152 are much less susceptible to cracking than Alloy 600 and its weld alloys. EPRI quotes a significant factor of improvement of Alloy 690 over Alloy 600 with similar FOI for the weld alloys [18]. It is noted though that weldments fabricated from Alloy 52 and 152 can be more prone to hot cracking during fabrications than those fabricated from Alloys 82 and 182.

The effect of cold work remains an open issue. Cold work has been identified as a potential cause of accelerated degradation in nickel base alloys and nickel base alloy welds. Within the NUGENIA+ project which was funded within the 7th Euratom framework programme of the European Commission, a mini-project, MICRIN+, experimentally surveyed the influence of the level of cold work on metallic surfaces on SCC crack initiation [19]. Subsequently a more extensive project, MEACTOS, funded under the EU Horizon 2020, NUGENIA+, Sustainable Nuclear Energy Technology Platform (SNETP) [20] is currently addressing the quantitative effects of surface working on environmental cracking of weld metal 182. This project has the objective mitigating environmental cracking of the nickel base alloy welds by optimizing surface processing. The project is specifically assessing the effects of surface machining practices on weld alloy 182 PWSCC crack initiation behavior [21].

1. ***Quality management:***

Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with the different national regulatory requirements (e.g., 10 CFR 50, Appendix B [22]).

**References**

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