## **AMP 105 BWR VESSEL ID ATTACHMENT WELDS (Version 2020)**

### Programme Description

The programme includes inspections and flaw evaluation to provide reasonable assurance of the long-term integrity and safe operation of BWR vessel inside diameter (ID) attachment welds. This programme provides inspection recommendations and evaluation methodologies for the attachment welds between the vessel wall and vessel ID brackets that attach safety-related components to the vessel (e.g., jet pump riser braces and core spray piping brackets). In some cases, the attachment is a simple weld; in others, it includes a weld build-up pad on the vessel.

### Evaluation and Technical Basis

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

The programme is focused on managing the effects of cracking due to stress corrosion cracking (SCC), including intergranular stress corrosion cracking (IGSCC) and irradiated assisted stress corrosion cracking (IASCC). The programme involves code-required examinations and/or an augmented in-service inspection programme that uses inspections and flaw evaluation to detect cracking and monitor the effects of cracking on the intended function of the components. The programme provides for repair and/or replacement, as needed, to maintain the ability to perform the intended function. The programme is applicable to structural welds for BWR reactor vessel internal integral attachments.

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

The BWR Vessel ID Attachment Welds Programme is a condition monitoring programme. Water chemistry control can reduce susceptibility to IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the plant water chemistry programme [1]. The programme description, evaluation, and technical basis of the water chemistry programme are presented in AMP 103.

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

The programme monitors for cracks induced by SCC on the intended function of BWR vessel ID attachment welds. The method, extent and schedule of the inspections are designed to maintain structural integrity and ensure that ageing effects are detected and managed to adequately maintain the intended functions of the component, consistent with pertinent governing requirements or guidance documents for the plant (e.g., [2-5]). Inspections can reveal cracking. Vessel ID attachment welds are inspected using visual VT-1 examination[[1]](#footnote-1) to detect discontinuities and imperfections on the surfaces of components and/or using visual VT-3 examination[[2]](#footnote-2) to determine the general mechanical and structural condition of the component supports.

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

This programme has no specific monitoring and trending activities. However, if flaws are detected, the scope of examinations is expanded, and flaw behavior is monitored by reinspection.

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

Water chemistry control can reduce susceptibility to IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the plant`s water chemistry programme. The programme description, evaluation, and technical basis are provided in AMP 103. Additional mitigation technologies can include surface treatment (peening and surface melting/solution annealing), as described in IAEA-TECDOC-1470 [6], Chapter 7, and OECD/NEA/CSNI/R (2010)15 [7].

* 1. ***Acceptance criteria:***

Acceptance criteria are provided by pertinent governing requirements or guidance documents for the plant. Examples of acceptance criteria are provided in:

1. Vessel ID attachment welds are inspected in accordance with the requirements of ASME Section XI [2], Subsection IWB, Examination Category B-N-2 which are specified in Table IWB-2500-1. The ASME Code, Section XI inspection specifies visual VT-1 examination to detect discontinuities and imperfections on the surfaces of components (e.g., cracks, wear, corrosion, and erosion) and visual VT-3 examination to determine the general mechanical and structural condition of the components. This programme looks for surface discontinuities that may indicate the presence of a crack. For a few locations (e.g. core spray piping brackets) the BWRVIP-48-A and BWRVIP-48 Revision 1 [3, 4] recommends an EVT-1 examination. Acceptance criteria can also be found in [3, 4]. Additional information for crack growth rates to use in evaluating cracking can be found in [8-14].
2. Inspection requirements specified in JSME S NA1 [5] table IB-2500-13, and acceptance criteria specified in EB-1200.
3. Additional guidance for treatment of flaws can be found in [14] which provides crack growth rates for irradiated stainless steel and is the basis for the upcoming ASME Code Case N-889.

In addition, IAEA-TECDOC-1470 [6] provides guidelines for the evaluation of detected cracks.

* 1. ***Corrective actions:***

Repair and replacement procedures are performed in accordance with pertinent governing requirements or guidance documents for the plant.

* 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.

Cracking due to IGSCC, has occurred in BWR components. The programme guidelines are based on an evaluation of available information, including BWR inspection data and information on the elements that cause IGSCC, to determine which attachment welds may be susceptible to cracking. Implementation of this programme provides reasonable assurance that cracking will be adequately managed and that the intended functions of the vessel ID attachments will be maintained consistent with the initial design basis for the period of extended operation. OECD/NEA/CSNI/R(2010)15 [7] has additional information on operating experience.

There are several international research and development programmes on IASCC and IGSCC, e.g. EPRI, US DoE, NUGENIA.

* 1. ***Quality management:***

Site quality assurance 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 [15]).

### References

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2. AMERICAN SOCIETY OF MECHANICAL ENGINEERS, Rules for Inservice Inspection of Nuclear Power Plant Components, The ASME Boiler and Pressure Vessel Code, ASME Section XI, as approved in 10 CFR 50.55a, ASME, New York, NY
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5. JAPAN SOCIETY OF MECHANICAL ENGINEERS, IA, IB Code for Nuclear Power Generation Facilities - Rule on Fitness-for-Service for Nuclear Power Plants, JSME S NA1 -2016, JSME
6. INTERNATIONAL ATOMIC ENERGY AGENCY, Assessment and management of ageing of major nuclear power plant components important to safety: BWR pressure vessels, IAEA-TECDOC-1470, IAEA, Vienna, October 2005
7. NUCLEAR ENERGY AGENCY, Technical Basis for Commendable Practices on Ageing Management-SCC and Cable Ageing Project (SCAP) Final Report, OECD/NEA/CSNI/ R(2010)15, NEA, Paris, April 2011
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11. AMERICAN SOCIETY OF MECHANICAL ENGINEERS, Code Case N-896, Reference Crack Growth Rate Curves for Stress, Corrosion Cracking of Low Alloy Steels in Boiling Water Reactor Environments Section XI, Division 1, August 2019
12. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-99-A: BWR Vessel and Internals Project, Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components, EPRI Technical Report 1016566, November 2008. Errata issued August 2002, BWRVIP letter 2002-219
13. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-100-A, Revision 1: BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds, EPRI Technical Report 3002008388, December 2016
14. ELECTRIC POWER RESEARCH INSTITUTE, Models of Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels in Light Water Reactor Environments EPRI, Palo Alto, CA: 2014. 3002003103
15. UNITED STATES NUCLEAR REGULATORY COMMISSION, 10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, USNRC, Latest Edition

1. IAEA-TECDOC-1470 [6]; Assessment and management of ageing of major nuclear power plant components important to safety: BWR pressure vessels; Chapter 5.2.3 ASME VT-1: a visual inspection method capable of achieving 0.8 mm resolution. VT-1 is conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion. [↑](#footnote-ref-1)
2. ASME VT-3: a visual inspection method for assessing the general mechanical and structural condition of components and their supports. Parameters such as clearances, settings, and physical displacements must be verified to detect discontinuities and imperfections such as loss of integrity at bolted or welded connections, loose or missing parts, corrosion, wear, or erosion. [↑](#footnote-ref-2)