## **AMP 108 BWR PENETRATIONS (Version 2020)**

### Programme Description

The programme for BWR vessel instrumentation penetrations, control rod drive housing (CRDH) and incore-monitoring housing (ICMH) penetrations and standby liquid control (SLC) nozzles/Core ΔP nozzles includes inspections and flaw evaluation. Although this is a condition monitoring programme, control of water chemistry helps prevent stress corrosion cracking (SCC), particularly intergranular stress corrosion cracking (IGSCC). The water chemistry programme for BWRs [1] relies on monitoring and control of reactor water chemistry and mitigates IGSCC of these reactor pressure vessel penetrations and nozzles. Adequate ageing management activities for these components provide reasonable assurance that the long-term integrity and safe operation of BWR vessel instrumentation nozzles, CRD housing and incore-monitoring housing (ICMH) penetrations and SLC nozzles/Core ΔP nozzles.

### Evaluation and Technical Basis

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

This programme is applicable to BWR instrumentation penetrations, CRDH and ICMH penetrations and BWR SLC nozzles/Core ΔP nozzles. The programme manages cracking due to cyclic loading or SCC and IGSCC using inspection and flaw evaluation.

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

This programme is a condition monitoring programme and has no preventive actions. However, water chemistry control can prevent or mitigate SCC, particularly IGSCC. The programme description, evaluation and technical basis are presented in AMP 103.

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

Examinations are implemented consistent with pertinent governing requirements of guidance documents for the plant [2-4]. IAEA-TECDOC-1470 [5], IAEA report NP-T-3.13 [6] and Nuclear Safety Standard KTA 3201.4 [7] provide guidance for monitoring the indications of cracking in BWR instrumentation nozzles, CRD housing and incore-monitoring housing (ICMH) penetrations, and BWR SLC nozzles/Core ΔP nozzles. These examinations include volumetric examination methods (ultrasonic testing or radiography testing), surface examination methods (liquid penetrant testing or magnetic particle testing), and VT-2 visual examination methods.

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

Inspections consistent with pertinent governing requirements or guidance documents for the plant can provide timely detection of cracks. The scope of examination and reinspection is expanded beyond the baseline inspections if flaws are detected. Any indication detected is evaluated in accordance with the pertinent governing requirements or guidance documents for the plant.

1. ***Mitigating ageing effect:***

Water chemistry control can reduce susceptibility to SCC, particularly IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Programme, as described in AMP 103. Mitigation technologies can include surface treatment (peening and surface melting / solution annealing), as described in IAEA-TECDOC-1470 [5], Section 7, and OECD/NEA/CSNI/R (2010)15 [8]. Mitigation is accomplished by eliminating at least two factors among three factors known to be necessary for stress corrosion cracking, susceptible material, significant tensile stress and a specific environment, especially when performing repair weld, mitigation, and replacement. Attention needs to be paid to the initiation of IGSCC, which has been observed at locations with high hardness due to grinding, or machining and weld residual stresses of the inside diameter near the HAZ.

* Material factors:

1. Select low-carbon grades of austenitic stainless steel, containing less carbon than specified in accordance with pertinent governing requirements or guidance documents for the plant.
2. A minimum ferrite of 7.5 % in weld metal and cast austenitic stainless steel (CASS).

* Residual stress factors:

1. Induction heating stress improvement (IHSI)
2. Peening (Shot peening, Water Jet Peening, Laser Peening, etc.)
3. Polishing
4. HSW (Heat Sink Welding)
5. Mechanical Stress Improvement (MSIP)

* Water chemistry factors

The applicable water chemistry options for mitigation are defined in AMP103. They include Hydrogen water chemistry (HWC), Noble Metal Chemical Application (NMCA) and Online Noble Metal Chemistry; all of which have been demonstrated to be effective, however may have limitations based on high fluence and locations.

1. ***Acceptance criteria:***

Acceptance criteria are provided by pertinent governing requirements [7, 9-10] or guidance documents for the plant. IAEA-TECDOC-1470 [5] provides guidelines for the evaluation of detected cracks. Additional information for evaluating cracking can be found in [11-16].

1. ***Corrective actions:***

Repair and replacement are performed in accordance with pertinent governing requirements or guidance documents for the plant. Additional guidance can be found in [17-20].

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 SCC, particularly IGSCC has occurred in BWR components made of austenitic SSs and nickel alloys. The programme guidelines are based on an evaluation of available information, including BWR inspection data and information about the elements that cause SCC, to determine which locations may be susceptible to cracking. Implementation of the programme provides reasonable assurance that cracking will be adequately managed so the intended functions of the BWR instrument penetrations, CRD housing and ICMH penetrations, and SLC/ΔP nozzles will be maintained consistent with the current licensing basis for the period of extended operation. Although there is limited inspection experience of SLC nozzle, one BWR (after 30 years operation) in Japan [21] conducted outside surface Visual Testing (VT) of a SLC nozzle and piping on the outside of the reactor vessel as a part of structure integrity confirmation activities after 3 years shutdown, during Ageing Management Technical Evaluation. As a result of this examination, no crack indication attributed to SCC was detected. However, as an additional evaluation to verify structure integrity, the utility conducted a residual stress analysis of weld, and concluded there is few possibility of SCC development based on the relationship between piping diameter and residual stress produced conservative result comparing with large bore piping evaluation, which is a result of past research.

During 2016 in the U.S., an ICMH flange weld outside the RPV was determined to have a through-wall leak [22]. The investigation revealed this weld had been repaired multiple times during fabrication. The location was repaired using a weld overlay.

In 2016 and 2017, two US BWRs have found indications in instrumentation penetrations resulting from stress corrosion cracking in the J-groove weld. In both cases, the cracking was detected visually through evidence of leakage and confirmed by UT. Cracking occurred in IGSCC susceptible Alloy 182 material. Half-nozzle repairs were implemented for both of these occurrences.

EPRI, CRIEPI and other research organizations have national and international research programmes on SCC for initiation and growth rates for stainless steels and nickel-based alloys, e.g. at EPRI [23].

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 [24].

### References

1. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-190 Revision 1: BWR Vessel and Internals Project, Volume 1: BWR Water Chemistry Guidelines – Mandatory, Needed, and Good Practice Guidance. EPRI, Palo Alto, CA: 2014. 3002002623
2. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-27-A: BWR Vessel and Internals Project, BWR Standby Liquid Control System/Core Plate Delta P Inspection and Flaw Evaluation Guidelines, EPRI Technical Report 1007279, August 2003
3. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-47-A: BWR Vessel and Internals Project, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines, EPRI Technical Report 1009947, November 2004
4. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-49-A: BWR Vessel and Internals Project, Instrument Penetration Inspection and Flaw Evaluation Guidelines, EPRI Technical Report 1006602, March 2002
5. 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
6. INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Nuclear Energy Series NP-T-3.13, Stress Corrosion Cracking in Light Water Reactors: Good Practices and Lessons Learned, 2011
7. KERNTECHNISCHER AUSSCHUSS (KTA), Components of the Reactor Coolant Pressure Boundary of Light Water Reactors, Part 4: Inservice Inspections and Operational Monitoring, Nuclear Safety Standard KTA 3201.4, November 2010
8. 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
9. 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
10. 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
11. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-14-A: BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals, EPRI Report 1016569, September 2008
12. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-59-A: BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Nickel Base Austenitic Alloys in RPV Internals, EPRI Technical Report 1014874, May 2007
13. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-60-A: BWR Vessel and Internals Project, Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment, EPRI Technical Report 1008871, June 2003
14. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-233, Revision 2: BWR Vessel and Internals Project: Updated Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment. EPRI, Palo Alto, CA: 2018. 3002013026
15. 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
16. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-100, Revision 1-A: BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds, EPRI Technical Report 3002008388, 2016
17. ELECTRIC POWER RESEARCH INSTITUTE, BWR Vessel and Internals Project, Roll/Expansion Repair of Control Rod Drive and In- Core Instrument Penetrations in BWR Vessels (BWRVIP-17), EPRI Report TR-106712, November 1996
18. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-53-A: BWR Vessel and Internals Project, Standby Liquid Control Line Repair Design Criteria, EPRI Technical Report 1012120, September 2005
19. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-57-A: BWR Vessel and Internals Project, Instrument Penetration Repair Design Criteria, EPRI Technical Report 1012111, September 2005
20. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-146NP, Revision 1: BWR Vessel and Internals Project, Technical Basis for ASME Code Case N-730, ‘Roll-Expansion of Class 1 Control Rod Drive Bottom Head Penetrations in BWRs,’ EPRI Technical Report 1016586, September 2008
21. Tokyo Electric Power Company Holdings, “Fukushima-daini Unit 3 Aging Management Technical Evaluation Report” (in Japanese), TEPCO, April 2015
22. UNITED STATES NUCLEAR REGULATORY COMMISSION, Licensee Event Report 50-387/2016-020-00, Reactor Coolant Pressure Boundary Leakage at LPRM Housing as a result of IGSCC, USNRC, August 3, 2016
23. ELECTRIC POWER RESEARCH INSTITUTE, Validation of Stress Corrosion Cracking Initiation Model for Stainless Steel and Nickel Alloys, EPRI Technical Report 1025121, December 2012
24. 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