## AMP 107 BWR Stress Corrosion Cracking in Coolant Pressure Boundary Components (Version 2020)

**Programme Description**

The main objective of this programme is to manage stress corrosion cracking (SCC), particularly intergranular stress corrosion cracking (IGSCC), in BWR coolant pressure boundary components made of stainless steel (SS) and nickel-based alloys, including welds. The programme includes (a) preventive measures to mitigate IGSCC and (b) inspec­tion and flaw evaluation to monitor IGSCC and its effects.

IGSCC is a form of SCC where cracking occurs along the grain boundaries. Stress corrosion cracking is a complex phenomenon driven by the synergistic interaction of mechanical, electrochemical, and metallurgical factors. Three preconditions are necessary and must be present simul­­tane­ously for SCC:

* A susceptible material, such as SS sensitised during welding;
* A significant tensile stress, such as residual stresses from fabrication processes;
* A corrosive environment, such as oxygenated hot water which contains higher contents of chlorides or sulphates.

Programmes and good practices to manage SCC in BWRs are delineated in various national and inter­national reports, such as:

* NUREG-0313, Rev. 2 [1], NRC Generic Letter (GL) 88 01 and its Supplement 1 [2], and BWRVIP-75-A [3] in the US;
* OECD NEA/CSNI/R (2010)15 [4];
* IAEA Technical Report No. NP-T-3.13 [5].

IAEA Technical Report NP-T-3.13 [5] is recommended as the reference document for this AMP, since it provides the most recent and most comprehensive international technical basis for managing SCC in light water reactors (LWRs). It delineates – among others – the state-of-the-art in science and technology regarding the mechanisms and major contri­buting factors to IGSCC in BWRs; the corresponding operating experiences; the ageing management application on IGSCC in BWRs including preventative actions to minimize and control ageing degradation, monitoring and trending of ageing effects, and acceptance criteria; inspection of components; and mitigation, repair, and replacement methods and strategies.

The EPRI Materials Degradation Matrix (MDM) [6] has a compilation of the materials used in the primary pressure boundary systems and components. It also identifies the modes of degradation that can affect the components and a high-level discussion of potential means to manage the mechanisms. The EPRI Materials Handbook [7] is broader in scope than the MDM and provides information for the primary pressure boundary and balance of plant components as well.

**Evaluation and Technical Basis**

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

The programme focuses on (a) managing and implementing countermeasures to mitigate IGSCC and (b) performing inservice inspections (ISI) to monitor IGSCC and its effects on the intended function of BWR components exposed to reactor coolant. The programme is applicable to all BWR piping and piping welds made of austenitic SS and nickel-based alloys that contain reactor coolant at a temperature above 100 degrees Celsius during power operation, regardless of code classification. The programme also applies to pump casings, valve bodies, and reactor vessel attachments and appurtenances, such as headspray and vent compo­nents. Control rod drive return line nozzle caps and associated welds may also be included in the scope of this programme. However, IGSCC of BWR vessel inside diameter attachment welds; BWR penetrations and nozzles; and BWR vessel internals is addressed in separate pro­grammes, mainly in AMP105, AMP108 and AMP109, respectively.

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

The elimination of any one of the three factors mentioned above (sensitive material, significant tensile stress, or corrosive environment) or the reduction of one of these three factors below some threshold level can typically prevent SCC. Therefore, the use of SCC resistant material, stress improvement processes, and water chemistry control can, in principle, prevent ageing degradation due to IGSCC. Since there are some uncertainties in the specific mechanism, elimination of only one of the three factors may not be sufficient. Therefore, it is recommended to eliminate at least two factors, if practical. This can be achieved by applying a combination of mitigation methods as outlined in Section 5 of this AMP.

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

The programme includes inspections that detect and size cracks and detect leakage by using examination and inspection methods consistent with the pertinent governing requirements or guidance documents for the plant. The equipment, personnel, and details of inspection, such as extent, method, and schedule of the inspections and test techniques, are based on the national codes and stan­dards and other pertinent governing requirements or guidance documents of each country, which are designed to maintain structural integrity. The codes, standards, and guidelines also ensure that ageing effects are detected and necessary corrective actions, such as sample size expansion to include additional components, repair and replacement, are conducted to adequately maintain the intended function of the component. The extent and frequency of inspections are based on the environmental conditions and/or as-fabricated conditions of each weld (e.g. whether hydrogen water chemistry is implemented and effective, the weldments were made from IGSCC-resistant material, whether a stress improvement process was applied to reduce residual stresses, and how the weld was repaired if it had cracked).

Detailed information on corresponding national inspection programmes is given in IAEA Technical Report No. NP-T-3.13. Accordingly, the inspections mainly rely on the following requirements as modified by the applicable guidance, e.g.:

* US NRC GL 88-01 [2];
* Section XI of the ASME Code [8] (e.g. USA, Spain, Switzerland);
* KTA 3201.4 [9] (e.g. Germany);
* JSME S NA1-2016 [10] and NISA-161a-03-01 (Japan) [11];
* SKIFS 2005:2 (Sweden; risk-based approach) [12];
* BWRVIP-75-A (EPRI 1012621), EPRI, Palo Alto, CA, October 2005 [13].

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

Monitoring and trending of IGSCC induced ageing includes detecting and sizing cracks, detecting reactor coolant leakage, analysing inspection results to ensure component integrity, and trending the operating experience. When one or more cracks are found, it is important to inspect an additional sample set of similar components. In addition, trending of the monitored electrochemical corrosion potential (ECP) is an effective method for this programme element, as well as monitoring and trending of the other water chemistry parameters.

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

Mitigating actions can be focussed on material, stress, or environmental aspects. Actions in the material context consist of the selection of SCC resistant materials, which include low carbon grades of austenitic SS with a maximum carbon content of 0.035 wt.%, and low carbon weld materials and cast austenitic stainless steels with a maximum carbon content of 0.035 wt.% and a minimum ferrite content of 7.5 % [2].

Mitigating actions regarding stress aspects include specially developed processes to relax residual tensile stresses, such as solution heat treatment (SHT), heat sink welding (HSW), induction heating stress improvement (IHSI), and mechanical stress improvement (MSIP). These processes are also designed to leave a compressive residual stress on the surface in contact with the reactor coolant. Such methods are described in detail in the Japanese “Guidelines for Preventive Maintenance Countermeasure” (e.g., [14-16]).

Methods for water chemistry control are established to control and monitor any adverse effects of the water chemistry conditions on the ageing effect. The pro­gramme description and evaluation and technical basis of monitoring and maintaining reactor coolant chemistry are addressed in AMP103. Further effective measures include hydrogen injection (i.e. hydro­gen water chemistry or HWC), noble metal technologies like noble metal chemical application (NMCA), and TiO2 injections [17-18]. However, in order to identify and assess any adverse effects on fuel performance and integrity or on radiation exposure resulting from changes to water chemistry, it is neces­sary to evaluate the latest observations and operating experience before implementation.

1. ***Acceptance criteria:***

Detected flaws are evaluated with the pertinent governing requirements or guidance documents for the plant. Corresponding procedures are described e.g. in IAEA Technical Report No. NP-T-3.13, Appendix II [5]. Preventive and mitigating actions may be credited in the evaluation only if its validity has been verified. In the case that the cracks are detected by the inspection, crack growth and fracture evaluation are conducted to confirm whether structural integrity can be maintained during further plant operations and for how long. Additional information for crack growth rates to use in evaluating cracking can be found in [19-20].

1. ***Corrective actions:***

Corrective actions include material changes, corrosion resistant cladding, weld material changes, design changes, weld overlays, stress improvements, environmental improve­ment, mechanical repair, and component replacement. Detailed information for these corrective actions is described in the guidelines such as IAEA Technical Report No. NP-T-3.13, NUREG-0313, Rev. 2 [5], ASME Section XI Code and Code Cases [8] and the 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.

BWR piping and piping components made of stabilized and non-stabilized austenitic SS or nickel-based alloys have experienced SCC and many cases have been reported throughout the world. The dominating early failure type in BWRs was IGSCC of sensitized SS and more recently IGSCC of cold worked SS (Type 316NG) [5, 21-22]. IGSCC has occurred for instance in recirculation, core spray, residual heat removal, CRD return line penetrations, and reactor water clean-up system piping welds.

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. UNITED STATES NUCLEAR REGULATORY COMMISSION, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, NUREG-0313, Rev. 2, USNRC, 1988
2. UNITED STATES NUCLEAR REGULATORY COMMISSION, NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, NRC Generic Letter 88-01, USNRC, January 25, 1988; Supplement 1, February 4, 1992
3. ELECTRIC POWER RESEARCH INSTITUTE, BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (NUREG-0313), BWRVIP-75-A (EPRI 1012621), EPRI, Palo Alto, CA, October 2005
4. NUCLEAR ENERGY AGENCY, Technical Basis for Commendable Practices on Ageing Management – SCC and Cable Ageing Project (SCAP), Final Report, NEA/CSNI/R (2010)15, NEA, Paris, April 2011
5. INTERNATIONAL ATOMIC ENERGY AGENCY, Stress Corrosion Cracking in Light Water Reactors: Good Practices and Lessons Learned, IAEA Nuclear Energy Series No. NP-T-3.13, IAEA, Vienna, 2011
6. ELECTRIC POWER RESEARCH INSTITUTE, EPRI Materials Degradation Matrix, Revision 4. EPRI, Palo Alto, CA: 2018. 3002013781
7. ELECTRIC POWER RESEARCH INSTITUTE, Materials Handbook for Nuclear Plant Pressure Boundary Applications (2018). EPRI, Palo Alto CA: 2018. 3002012420
8. AMERICAN SOCIETY of MECHANICAL ENGINEERS, ASME Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, The ASME Boiler and Pressure Vessel Code, as approved in 10 CFR50.55a, The American Society of Mechanical Engineers, New York, NY
9. KERNTECHNISCHER AUSSCHUSS, Components of the Reactor Coolant Pressure Boundary of Light Water Reactors, Part 4: Inservice Inspections and Operational Monitoring, Nuclear Safety Standard KTA 3201.4, KTA, November 2010
10. JAPAN SOCIETY OF MECHANICAL ENGINEERS, NA1, IA, IB Code for Nuclear Power Generation Facilities, Rule on Fitness-for-Service for Nuclear Power Plants, JSME S NA1 -2016, JSME
11. Nuclear and Industrial Safety Agency, “Inspection of Crack in Core Shroud and Primary Loop Recirculation Piping, etc.” (in Japanese), NISA-161a-03-01, NISA, April 2003
12. Swedish Radiation Safety Authority, Strålsäkerhetsmyndighetens föreskrifter om mekaniska anordningar i vissa kärntekniska anläggningar; SSMFS 2008:13, ISSN 2000-0987, 2008
13. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-75-A: BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules, EPRI Technical Report 1012621, October 2005
14. JAPAN NUCLEAR TECHNOLOGY INSTITUTE, Guideline for Preventive Maintenance Countermeasure, Stress Improvement Method by Outer Surface Heating, JANTI-VIP-02, JANTI
15. JAPAN NUCLEAR TECHNOLOGY INSTITUTE, Guideline for Preventive Maintenance Countermeasure, Peening, JANTI-VIP-03, JANTI
16. JAPAN NUCLEAR TECHNOLOGY INSTITUTE, Guideline for Preventive Maintenance Countermeasure, Stress Improvement by Polishing, JANTI-VIP-10, JANTI
17. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-62, Revision 1: BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection, EPRI Technical Report 1022844, December 2011
18. ELECTRIC POWER RESEARCH INSTITUTE, BWRVIP-190 Revision 1: BWR Vessel and Internals Project, BWR Water Chemistry Guidelines - 2008 Revision, EPRI Technical Report 3002002623, April 2014
19. 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
20. 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
21. N. Ishiyama et al.: Stress Corrosion Cracking of Type 316 and 316L Stainless Steels in High Temperature Water, Proceedings of the 12th International Conference on Environmental Degradation of Materials in Nuclear Power System – Water Reactors, 2005
22. S. Suzuki, et al.: Stress Corrosion Cracking in Low Carbon Stainless Steel Com­ponents in BWRs, E-Journal of Advanced Maintenance, Vol. 1 (2009) 1-29, Japan Society of Maintenology
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