## AMP 129 BWR Reactor Water Cleanup System (VERSION 2021)

### Programme Description

This programme provides inspection to manage the ageing effects of cracking due to stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) on the intended function of austenitic stainless steel (SS) piping outboard of the second primary containment isolation valves or where the Class 1 piping boundary ends in the reactor water cleanup (RWCU) system. For this system, the programme covers all BWR piping made of austenitic SS that is 100 mm (4 inches) or larger in nominal diameter and contains reactor coolant at a temperature above 93.3 °C (200 °F) during power operation regardless of classification.

### Evaluation and Technical Basis

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

The AMP monitors SCC or IGSCC of austenitic SS piping by detecting and sizing cracks in accordance with the national requirements. It focuses on performing in-service inspections to monitor the cracking due to SCC or IGSCC in austenitic SS piping outboard of the second containment isolation valves in the RWCU system. The components included in this programme are the welds in piping that have a nominal diameter of 100 mm (4 inches) or larger and that contain reactor coolant at a temperature above 93.3 °C (200 °F) during power operation, regardless of code classification.

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

As described in AMP 107 and in [1] in more detail, the elimination of any one of the three factors:

* Sensitive material;
* Significant tensile stress;
* Corrosive environment.

or the reduction of one of these three factors below some threshold level can prevent or retard SCC/IGSCC. Therefore, the use of SCC/IGSCC resistant material, stress improvement processes and water chemistry control can, in principle, prevent ageing degradation due to SCC. Since there are some uncertainties in the specific mechanism; elimination of only one of the three factors may not be sufficient.

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

The extent, method, and schedule of the inspection and test techniques are sufficient to maintain structural integrity and to detect ageing effects before the loss of intended function of austenitic SS piping and fittings [2-3]. Guidelines for the inspection schedule, methods, personnel, sample expansion, and leak detection are based on national requirements, and would depend on susceptibility of the piping material to IGSCC and prior inspection findings. For example, in the United States according to [4, 5], page XI M25-2, no inspection of the outboard piping is required for:

* Piping systems that are made of IGSCC-resistant piping materials (e.g., carbon content of the stainless steel is < 0.035 %, and the delta ferrite in the weld metal is a minimum 7.5 %) [2];
* Piping with no IGSCC detected inboard of the second isolation valves and outboard of the second isolation valves (after inspecting a minimum of 10 % of susceptible piping welds).

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

The inspection provides timely detection of cracks and leakage of coolant. Based on inspection results, additional samples of welds are inspected when one or more cracked welds are found in a weld category.

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

As described in AMP 107 and in [1, 6] in more detail, mitigating actions can be focussed on material, stress or environmental aspects. Methods for water chemistry control are established to control and monitor any adverse effects of the water chemistry conditions on the ageing effect (e.g. [7]). The programme description and evaluation and technical basis of monitoring and maintaining reactor coolant chemistry are addressed in AMP 103.

1. ***Acceptance criteria:***

Detected flaws are evaluated with the pertinent governing requirements or guidance documents for the plant [8-11]. Corresponding procedures are described e.g. in IAEA Technical Report No. NP-T-3.13, Appendix II [3]. 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.

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 NUREG-0313, Rev. 2 [12] and IAEA Technical Report No. NP-T-3.13 [6] 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.

The IGSCC has occurred in small- and large-diameter BWR piping made of austenitic stainless steels. IGSCC has occurred for instance in recirculation, core spray, residual heat removal, CRD return line penetrations, and reactor water cleanup system piping welds.

EPRI, CRIEPI and other international research organizations have on-going research programmes for SCC initiation and growth rate.

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 [13]).

### References

1. ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT, NEA/CSNI/R(2010)15, Technical Basis for Commendable Practices on Ageing Management – SCC and Cable Ageing Project (SCAP), OECD, Final Report, April 2011.
2. UNITED STATES NUCLEAR REGULATORY COMMISSION, Generic Letter 88-01, NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, U.S. Nuclear Regulatory Commission, January 25, 1988; Supplement 1, February 4, 1992.
3. 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.
4. UNITED STATES NUCLEAR REGULATORY COMMISSION, Generic Aging Lessons Learned (GALL) Report — Final Report (NUREG-1801, Revision 2), USNRC, 2010.
5. UNITED STATES NUCLEAR REGULATORY COMMISSION, Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report — Final Report (NUREG-2191, Vol. 2), USNRC, 2017.
6. INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Nuclear Energy Series No. NP-T-3.13, Stress Corrosion Cracking in Light Water Reactors: Good Practices and Lessons Learned, IAEA, Vienna, 2011.
7. 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 Report 3002002623, April 2014.
8. 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.
9. 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, 2012 (Addendum 2015) , JSME.
10. 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.
11. 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.
12. UNITED STATES NUCLEAR REGULATORY COMMISSION, NUREG-0313, Rev. 2, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, W. S. Hazelton and W. H. Koo, USNRC 1988.
13. UNITED STATES NUCLEAR REGULATORY COMMISSION, Title 10 Part 50 of the Code of Federal Regulations (10 CFR 50), Appendix B, Quality Assurance Criteria for Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, USNRC, Latest Edition.