XI.M7 BWR STRESS CORROSION CRACKING

Program Description

The program to manage intergranular stress corrosion cracking (IGSCC) in boiling water reactor (BWR) coolant pressure boundary piping made of stainless steel (SS) and nickel-based alloy components is delineated in NUREG-0313, Rev. 2, and Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01 and its Supplement 1. The material includes base metal and welds. The comprehensive program outlined in NUREG-0313, Rev 2 and NRC GL 88-01 describes improvements that, in combination, will reduce the susceptibility to IGSCC. The elements to cause IGSCC consist of a susceptible (sensitized) material, a significant tensile stress, and an aggressive environment. Sensitization of nonstabilized austenitic stainless steels containing greater than 0.035 weight percent carbon involves precipitation of chromium carbides at the grain boundaries during certain fabrication or welding processes. The formation of carbides creates a chromium-depleted region that, in certain environments, is susceptible to stress corrosion cracking (SCC). Residual tensile stresses are introduced from fabrication processes, such as welding, surface grinding, or forming. High levels of dissolved oxygen or aggressive contaminants, such as sulfates or chlorides, accelerate the SCC processes. The program includes (a) preventive measures to mitigate IGSCC and (b) inspection and flaw evaluation to monitor IGSCC and its effects. The staff-approved boiling water reactor vessel and internals project (BWRVIP-75-A) report allows for modifications to the inspection extent and schedule described in the GL 88-01 program.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program focuses on (a) managing and implementing countermeasures to mitigate IGSCC and (b) performing in-service inspection to monitor IGSCC and its effects on the intended function of BWR piping components within the scope of license renewal. The program is applicable to all BWR piping and piping welds made of austenitic SS and nickel alloy that are 4 inches or larger in nominal diameter containing reactor coolant at a temperature above 93°C (200°F) during power operation, regardless of code classification. The program also applies to pump casings, valve bodies, and reactor vessel attachments and appurtenances, such as head spray and vent components. NUREG-0313, Rev. 2 and NRC GL 88-01, respectively, describe the technical basis and staff guidance regarding mitigation of IGSCC in BWRs. Attachment A of NRC GL 88-01 delineates the staff-approved positions regarding materials, processes, water chemistry, weld overlay reinforcement, partial replacement, stress improvement of cracked welds, clamping devices, crack characterization and repair criteria, inspection methods and personnel, inspection schedules, sample expansion, leakage detection, and reporting requirements.

2. ***Preventive Actions:*** The BWR Stress Corrosion Cracking Program is primarily a condition monitoring program. Maintaining high water purity reduces susceptibility to SCC or IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description, evaluation and technical basis of water chemistry are addressed through implementation of GALL AMP XI.M2, “Water Chemistry.” In addition, NUREG-0313, Rev. 2 and GL 88-01 delineate the guidance for selection of resistant materials and processes that provide resistance to IGSCC such as solution heat treatment and stress improvement processes.

3. ***Parameters Monitored/Inspected:*** The program detects and sizes cracks and detects leakage by using the examination and inspection guidelines delineated in NUREG-0313, Rev. 2, and NRC GL 88-01 or the referenced BWRVIP-75-A guideline as approved by the NRC staff.

4. ***Detection of Aging Effects:*** The extent, method, and schedule of the inspection and test techniques delineated in NRC GL 88-01 or BWRVIP-75-A are designed to maintain structural integrity and ensure that aging effects are discovered and repaired before the loss of intended function of the component. Modifications to the extent and schedule of inspection in NRC GL 88-01 are allowed in accordance with the inspection guidance in approved BWRVIP-75-A. The program uses volumetric examinations to detect IGSCC. Inspection can reveal cracking and leakage of coolant. The extent and frequency of inspection recommended by the program are based on the condition of each weld (e.g., whether the weldments were made from IGSCC-resistant material, whether a stress improvement process was applied to a weldment to reduce residual stresses, and how the weld was repaired, if it had been cracked).

5. ***Monitoring and Trending:*** The extent and schedule for inspection, in accordance with the recommendations of NRC GL 88-01 or approved BWRVIP-75-A guidelines, provide timely detection of cracks and leakage of coolant. Indications of cracking are evaluated and trended in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, IWA-3000.

   Applicable and approved BWRVIP-14-A, BWRVIP-59-A, BWRVIP-60-A, and BWRVIP-62 reports provide guidelines for evaluation of crack growth in SSs, nickel alloys, and low-alloy steels. An applicant may use BWRVIP-61 guidelines for BWR vessel and internals induction heating stress improvement effectiveness on crack growth in operating plants.

6. ***Acceptance Criteria:*** Any cracking is evaluated in accordance with ASME Code, Section XI, IWA-3000 by comparing inspection results with the acceptance standards of ASME Code, Section XI, IWB-3000, IWC-3000 and IWD-3000 for Class 1, 2 and 3 components, respectively.

7. ***Corrective Actions***: The guidance for weld overlay repair and stress improvement or replacement is provided in NRC GL 88-01. Corrective action is performed in accordance with IWA-4000. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the administrative controls.

10. ***Operating Experience:*** Intergranular SCC has occurred in small- and large-diameter BWR piping made of austenitic SS and nickel-base alloys. Cracking has occurred in recirculation, core spray, residual heat removal, CRD return line penetrations, and reactor water cleanup

system piping welds (NRC GL 88-01 and NRC Information Notices [INs] 82-39, 84-41, and 04-08). The comprehensive program outlined in NRC GL 88-01, NUREG-0313, Rev. 2, and in the staff-approved BWRVIP-75-A report addresses mitigating measures for SCC or IGSCC (e.g., susceptible material, significant tensile stress, and an aggressive environment). The GL 88-01 program, with or without the modifications allowed by the staff-approved BWRVIP-75-A report, has been effective in managing IGSCC in BWR reactor coolant pressure-retaining components and will adequately manage IGSCC degradation.

References

10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2009.

ASME Code Case N-504-1, *Alternative Rules for Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping*, Section XI, Division 1, 1995 edition, ASME Boiler and Pressure Vessel Code – Code Cases – Nuclear Components, American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, The American Society of Mechanical Engineers, New York, NY.

BWRVIP-14-A (EPRI 1016569), *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, September 2008.

BWRVIP-59-A, (EPRI 1014874), *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals*, Final Report by the Office of Nuclear Reactor Regulation, May 2007.

BWRVIP-60-A (EPRI 108871), *BWR Vessel and Internals Project, Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, June 2003.

BWRVIP-61 (EPRI 112076), *BWR Vessel and Internals Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Reactors,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, January 29, 1999.

BWRVIP-62 (EPRI 108705), *BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection,* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, March 7, 2000.

BWRVIP-75-A (EPRI 1012621), *BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (NUREG-0313),* Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation, October 2005.

NRC 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.

NRC Information Notice 04-08, *Reactor Coolant Pressure Boundary Leakage Attributable to Propagation of Cracking in Reactor Vessel Nozzle Welds,* U.S. Nuclear Regulatory Commission, April 22, 2004.

NRC Information Notice 82-39, *Service Degradation of Thick Wall Stainless Steel Recirculation System Piping at a BWR Plant*, U.S. Nuclear Regulatory Commission, September 21, 1982.

NRC Information Notice 84-41, *IGSCC in BWR Plants*, U.S. Nuclear Regulatory Commission, June 1, 1984.

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, U.S. Nuclear Regulatory Commission, 1988.