XI.M3 REACTOR HEAD CLOSURE STUD BOLTING

Program Description

This program includes (a) inservice inspection (ISI) in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB (2004 edition,[2] no addenda), Table IWB 2500-1; and (b) preventive measures to mitigate cracking. The program also relies on recommendations to address reactor head stud bolting degradation as delineated in NUREG-1339 and Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.65.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program manages the aging effects of cracking due to stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) and loss of material due to wear or corrosion for reactor vessel closure stud bolting (studs, washers, bushings, nuts, and threads in flange) for both boiling water reactors (BWRs) and pressurized water reactors (PWRs).

2. ***Preventive Actions:*** Preventive measures include:

   (a) avoiding the use of metal-plated stud bolting to prevent degradation due to corrosion or hydrogen embrittlement;

   (b) using manganese phosphate or other acceptable surface treatments;

   (c) using stable lubricants. Of particular note, use of molybdenum disulfide (MoS_2) as a lubricant has been shown to be a potential contributor to SCC and should not be used (RG 1.65); and

   (d) using bolting material for closure studs that has an actual measured yield strength less than 1,034 megapascals (MPa) (150 kilo-pounds per square inch) (NUREG-1339).

   Implementation of these mitigation measures can reduce potential for SCC or IGSCC, thus making this program effective.

3. ***Parameters Monitored/Inspected:*** The ASME Section XI ISI program detects and sizes cracks, detects loss of material, and detects coolant leakage by following the examination and inspection requirements specified in Table IWB-2500-1.

4. ***Detection of Aging Effects:*** The extent and schedule of the inspection and test techniques prescribed by the program are designed to maintain structural integrity and ensure that aging effects are discovered and repaired before the loss of intended function of the component. Inspection can reveal cracking, loss of material due to corrosion or wear, and leakage of coolant.

   The program uses visual, surface, and volumetric examinations in accordance with the general requirements of Subsection IWA-2000. Surface examination uses magnetic particle or liquid penetrant examinations to indicate the presence of surface discontinuities and

[2] Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

flaws. Volumetric examination uses radiographic or ultrasonic examinations to indicate the presence of discontinuities or flaws throughout the volume of material. Visual VT-2 examination detects evidence of leakage from pressure-retaining components, as required during the system pressure test.

Components are examined and tested in accordance with ASME Code, Section XI, Table IWB-2500-1, Examination Category B-G-1, for pressure-retaining bolting greater than 2 inches in diameter. Examination Category B-P for all pressure-retaining components specifies visual VT-2 examination of all pressure-retaining boundary components during the system leakage test. Table IWB-2500-1 specifies the extent and frequency of the inspection and examination methods, and IWB-2400 specifies the schedule of the inspection.

5. ***Monitoring and Trending:*** The Inspection schedule of IWB-2400 and the extent and frequency of IWB-2500-1 provide timely detection of cracks, loss of material, and leakage.

6. ***Acceptance Criteria:*** Any indication or relevant condition of degradation in closure stud bolting is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500.

7. ***Corrective Actions:*** Repair and replacement are performed in accordance with the requirements of IWA-4000 and the material and inspection guidance of RG 1.65. The maximum yield strength of replacement material should be limited as recommended in NUREG-1339. 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:*** SCC has occurred in BWR pressure vessel head studs (Stoller, 1991). The aging management program has provisions regarding inspection techniques and evaluation, material specifications, corrosion prevention, and other aspects of reactor pressure vessel head stud cracking. Implementation of the program provides reasonable assurance that the effects of cracking due to SCC or IGSCC and loss of material due to wear are adequately managed so that the intended functions of the reactor head closure studs and bolts are maintained consistent with the current licensing basis for the period of extended operation. Degradation of threaded bolting and fasteners in closures for the reactor coolant pressure boundary has occurred from boric acid corrosion, SCC, and fatigue loading (NRC Inspection and Enforcement Bulletin 82-02, NRC Generic Letter 91-17).

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

NRC Regulatory Guide 1.65, *Material and Inspection for Reactor Vessel Closure Studs*, Revision 1, U.S. Nuclear Regulatory Commission, April 2010.

NRC Inspection and Enforcement Bulletin 82-02, *Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants*, June 2, 1982.

NUREG-1339, *Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants*, June 1990.

NRC Generic Letter 91-17, *Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants,* October 17, 1991.

Stoller, S. M., Reactor Head Closure Stud Cracking, Material Toughness Outside FSAR - SCC in Thread Roots, Nuclear Power Experience, BWR-2, III, 58, p. 30, 1991.