

XI.M1 ASME SECTION XI INSERVICE INSPECTION, SUBSECTIONS IWB, IWC, AND IWD

Program Description

Title 10 of the *Code of Federal Regulations*, 10 CFR 50.55a, imposes the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, for Class 1, 2, and 3 pressure-retaining components and their integral attachments in light-water cooled power plants. Inspection of these components is covered in Subsections IWB, IWC, and IWD, respectively, in the 2004 edition.¹ The program generally includes periodic visual, surface, and/or volumetric examination and leakage test of all Class 1, 2, and 3 pressure-retaining components and their integral attachments. Repair/replacement activities for these components are covered in Subsection IWA of the ASME code.

The ASME Section XI inservice inspection program, in accordance with Subsections IWB, IWC, or IWD, has been shown to be generally effective in managing aging effects in Class 1, 2, or 3 components and their integral attachments in light-water cooled power plants. 10 CFR 50.55a imposes additional limitations, modifications, and augmentations of ISI requirements specified in ASME Code, Section XI, and those limitations, modifications, or augmentations described in 10 CFR 50.55a are included as part of this program. In certain cases, the ASME inservice inspection program is to be augmented to manage effects of aging for license renewal and is so identified in the Generic Aging Lessons Learned (GALL) Report.

Evaluation and Technical Basis

1. **Scope of Program:** The ASME Section XI program provides the requirements for ISI, repair, and replacement of code Class 1, 2, or 3 pressure-retaining components and their integral attachments in light-water cooled nuclear power plants. The components within the scope of the program are specified in ASME Code, Section XI, Subsections IWB-1100, IWC-1100, and IWD-1100 for Class 1, 2, and 3 components, respectively. The components described in Subsections IWB-1220, IWC-1220, and IWD-1220 are exempt from the volumetric and surface examination requirements, but not exempt from visual exam requirements of Subsections IWB-2500, IWC-2500, and IWD-2500.
2. **Preventive Actions:** This is a condition monitoring program. It does not implement preventive actions.
3. **Parameters Monitored/Inspected:** The ASME Section XI ISI program detects degradation of components by using the examination and inspection requirements specified in ASME Section XI Tables IWB-2500-1, IWC-2500-1, or IWD-2500-1, respectively, for Class 1, 2, or 3 components.

The program uses three types of examination—visual, surface, and volumetric—in accordance with the requirements of Subsection IWA-2000. Visual VT-1 examination detects discontinuities and imperfections, such as cracks, corrosion, wear, or erosion, on the surface of components. Visual VT-2 examination detects evidence of leakage from pressure-retaining components, as required during the system pressure test. Visual VT-3 examination (a) determines the general mechanical and structural condition of components and their supports by verifying parameters such as clearances, settings, and physical

¹ Refer to the GALL Report, Chapter I, for applicability of other editions of the ASME Code, Section XI.

displacements; (b) detects discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion; and (c) observes conditions that could affect operability or functional adequacy of constant-load and spring-type components and supports.

Surface examination uses magnetic particle, liquid penetrant, or eddy current examinations to indicate the presence of surface discontinuities and flaws.

Volumetric examination uses radiographic, ultrasonic, or eddy current examinations to indicate the presence of discontinuities or flaws throughout the volume of material included in the inspection program.

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, leakage of coolant, and indications of degradation due to wear or stress relaxation (such as changes in clearances, settings, physical displacements, loose or missing parts, debris, wear, erosion, or loss of integrity at bolted or welded connections).

Components are examined and tested as specified in Tables IWB-2500-1, IWC-2500-1, and IWD-2500-1, respectively, for Class 1, 2, and 3 components. The tables specify the extent and schedule of the inspection and examination methods for the components of the pressure-retaining boundaries. Alternative approved methods that meet the requirements of IWA-2240 are also specified in these tables. For boiling water reactors (BWRs), the nondestructive examination (NDE) techniques appropriate for inspection of vessel internals, including the uncertainties inherent in delivering and executing an NDE technique in a BWR, are included in the approved Boiling Water Reactor Vessel and Internals Project Report (BWRVIP-03).

5. **Monitoring and Trending:** For Class 1, 2, or 3 components, the inspection schedule of IWB-2400, IWC-2400, or IWD-2400, respectively, and the extent and frequency of IWB-2500-1, IWC-2500-1, or IWD-2500-1, respectively, provides for timely detection of degradation. The sequence of component examinations established during the first inspection interval is repeated during each successive inspection interval, to the extent practical. If flaw conditions or relevant conditions of degradation are evaluated in accordance with IWB-3100, IWC-3100, or IWD-3000 and the component is qualified as acceptable for continued service, the areas containing such flaw indications and relevant conditions are reexamined during the next three inspection periods of IWB-2410 for Class 1 components, IWC-2410 for Class 2 components, and IWD-2410 for Class 3 components. Examinations that reveal indications that exceed the acceptance standards described below are extended to include additional examinations in accordance with IWB-2430, IWC-2430, or IWD-2430 for Class 1, 2, or 3 components, respectively.
6. **Acceptance Criteria:** Any indication or relevant conditions of degradation are evaluated in accordance with IWB-3000, IWC-3000, or IWD-3000 for Class 1, 2, or 3 components, respectively. Examination results are evaluated in accordance with IWB-3100 or IWC-3100 by comparing the results with the acceptance standards of IWB-3400 and IWB-3500, or IWC-3400 and IWC-3500, respectively, for Class 1 or Class 2 and 3 components. Flaws that exceed the size of allowable flaws, as defined in IWB-3500 or IWC-3500, are evaluated by using the analytical procedures of IWB-3600 or IWC-3600, respectively, for Class 1 or Class

2 and 3 components. Flaws that exceed the size of allowable flaws, as defined in IWB-3500 or IWC-3500, are evaluated by using the analytical procedures of IWB-3600 or IWC-3600, respectively, for Class 1 or Class 2 and 3 components.

7. **Corrective Actions:** Repair and replacement activities are performed in conformance 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:** Because the ASME Code is a consensus document that has been widely used over a long period, it has been shown to be generally effective in managing aging effects in Class 1, 2, and 3 components and their integral attachments in light-water cooled power plants (see Chapter I of the GALL Report).

Some specific examples of operating experience of component degradation are as follows:

BWR: Cracking due to intergranular stress corrosion cracking (IGSCC) has occurred in small- and large-diameter BWR piping made of austenitic stainless steels and nickel alloys. IGSCC has also occurred in a number of vessel internal components, such as core shrouds, access hole covers, top guides, and core spray spargers (U.S. Nuclear Regulatory Commission [NRC] Bulletin 80-13, NRC Information Notice [IN] 95-17, NRC Generic Letter [GL] 94-03, and NUREG-1544). Cracking due to thermal and mechanical loading has occurred in high-pressure coolant injection piping (NRC IN 89-80) and instrument lines (NRC Licensee Event Report [LER] 50-249/99-003-01). Jet pump BWRs are designed with access holes in the shroud support plate at the bottom of the annulus between the core shroud and the reactor vessel wall. These holes are used for access during construction and are subsequently closed by welding a plate over the hole. Both circumferential (NRC IN 88-03) and radial cracking (NRC IN 92-57) have been observed in access hole covers. Failure of the isolation condenser tube bundles due to thermal fatigue and transgranular stress corrosion cracking (TGSCC) caused by leaky valves has also occurred (NRC LER 50-219/98-014-00).

PWR Primary System: Although the primary pressure boundary piping of PWRs has generally not been found to be affected by stress corrosion cracking (SCC) because of low dissolved oxygen levels and control of primary water chemistry, SCC has occurred in safety injection lines (NRC IN 97-19 and 84-18), charging pump casing cladding (NRC IN 80-38 and 94-63), instrument nozzles in safety injection tanks (NRC IN 91-05), control rod drive seal housing (NRC Inspection Report 50-255/99012), and safety-related stainless steel (SS) piping systems that contain oxygenated, stagnant, or essentially stagnant borated coolant (NRC IN 97-19). Cracking has occurred in SS baffle former bolts in a number of foreign plants (NRC IN 98-11) and has been observed in plants in the United States. Cracking due to thermal and mechanical loading has occurred in high-pressure injection and safety

injection piping (NRC IN 97-46 and NRC Bulletin 88-08). Through-wall circumferential cracking has been found in reactor pressure vessel head control rod drive penetration nozzles (NRC IN 2001-05). Evidence of reactor coolant leakage, together with crack-like indications, has been found in bottom-mounted instrumentation nozzles (NRC IN 2003-11 and IN 2003-11, Supplement 1). Cracking in pressurizer safety and relief line nozzles and in surge line nozzles has been detected (NRC IN 2004-11), and circumferential cracking in stainless steel pressurizer heater sleeves has also been found (NRC IN 2006-27). Also, primary water stress corrosion cracking (PWSCC) has been observed in steam generator drain bowl welds inspected as part of a licensee's Alloy 600/82/182 program (NRC IN 2005-02).

PWR Secondary System: Steam generator tubes have experienced outside diameter stress corrosion cracking (OGSCC), intergranular attack, wastage, and pitting (NRC IN 97-88). Carbon steel support plates in steam generators have experienced general corrosion. Steam generator shells have experienced pitting and stress corrosion cracking (NRC INs 82-37, 85-65, and 90-04).

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.
- BWRVIP-03, *BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines (EPRI TR-105696 R1, March 30, 1999)*, Final Safety Evaluation Report by the Office of Nuclear Reactor Regulation for BWRVIP-03, July 15, 1999.
- NRC Bulletin 88-08, *Thermal Stresses in Piping Connected to Reactor Coolant System*, U.S. Nuclear Regulatory Commission, June 22, 1988; Supplement 1, June 24, 1988; Supplement 2, September 4, 1988; Supplement 3, April 4, 1989.
- NRC Generic Letter 94-03, *Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors*, U.S. Nuclear Regulatory Commission, July 25, 1994.
- NRC Bulletin 80-13, *Cracking in Core Spray Spargers*, U.S. Nuclear Regulatory Commission, May 12, 1980.
- NRC Information Notice 80-38, *Cracking in Charging Pump Casing Cladding*, U.S. Nuclear Regulatory Commission, October 31, 1980.
- NRC Information Notice 82-37, *Cracking in the Upper Shell to Transition Cone Girth Weld of a Steam Generator at an Operating PWR*, U.S. Nuclear Regulatory Commission, September 16, 1982.

NRC Information Notice 84-18, *Stress Corrosion Cracking in PWR Systems*, U.S. Nuclear Regulatory Commission, March 7, 1984.

NRC Information Notice 85-65, *Crack Growth in Steam Generator Girth Welds*, U.S. Nuclear Regulatory Commission, July 31, 1985.

NRC Information Notice 88-03, *Cracks in Shroud Support Access Hole Cover Welds*, U.S. Nuclear Regulatory Commission, February 2, 1988.

NRC Information Notice 89-80, *Potential for Water Hammer, Thermal Stratification, and Steam Binding in High-Pressure Coolant Injection Piping*, U.S. Nuclear Regulatory Commission, December 1, 1989.

NRC Information Notice 90-04, *Cracking of the Upper Shell-to-Transition Cone Girth Welds in Steam Generators*, U.S. Nuclear Regulatory Commission, January 26, 1990.

NRC Information Notice 91-05, *Intergranular Stress Corrosion Cracking in Pressurized Water Reactor Safety Injection Accumulator Nozzles*, U.S. Nuclear Regulatory Commission, January 30, 1991.

NRC Information Notice 92-57, *Radial Cracking of Shroud Support Access Hole Cover Welds*, U.S. Nuclear Regulatory Commission, August 11, 1992.

NRC Information Notice 94-63, *Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks*, U.S. Nuclear Regulatory Commission, August 30, 1994.

NRC Information Notice 95-17, *Reactor Vessel Top Guide and Core Plate Cracking*, U.S. Nuclear Regulatory Commission, March 10, 1995.

NRC Information Notice 97-19, *Safety Injection System Weld Flaw at Sequoyah Nuclear Power Plant, Unit 2*, U.S. Nuclear Regulatory Commission, April 18, 1997.

NRC Information Notice 97-46, *Unisolable Crack in High-Pressure Injection Piping*, U.S. Nuclear Regulatory Commission, July 9, 1997.

NRC Information Notice 97-88, *Experiences During Recent Steam Generator Inspections*, U.S. Nuclear Regulatory Commission, December 16, 1997.

NRC Information Notice 98-11, *Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants*, U.S. Nuclear Regulatory Commission, March 25, 1998.

NRC Information Notice 2001-05, *Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3*, U.S. Nuclear Regulatory Commission, April 30, 2001.

NRC Information Notice 2003-11, *Leakage Found on Bottom-Mounted Instrumentation Nozzles*, U.S. Nuclear Regulatory Commission, August 13, 2003.

NRC Information Notice 2003-11, Supplement 1, *Leakage Found on Bottom-Mounted Instrumentation Nozzles*, U.S. Nuclear Regulatory Commission, January 8, 2004.

NRC Information Notice 2004-11, *Cracking in Pressurizer Safety and Relief Nozzles and in Surge Line Nozzles*, U.S. Nuclear Regulatory Commission, May 4, 2004.

NRC Information Notice 2005-02, *Pressure Boundary Leakage Identified on Steam Generator Drain Bowl Welds*, U.S. Nuclear Regulatory Commission, February 4, 2005.

NRC Information Notice 2006-27, *Circumferential Cracking in the Stainless Steel Pressurizer Heater Sleeves of Pressurized Water Reactors*, U.S. Nuclear Regulatory Commission, December 11, 2006.

NRC Inspection Report 50-255/99012, *Palisades Inspection Report*, Item E8.2, Licensee Event Report 50-255/99-004, *Control Rod Drive Seal Housing Leaks and Crack Indications*, U.S. Nuclear Regulatory Commission, January 12, 2000.

NRC Licensee Event Report LER 50-219/98-014-00, *Failure of the Isolation Condenser Tube Bundles due to Thermal Stresses/Transgranular Stress Corrosion Cracking Caused by Leaky Valve*, U.S. Nuclear Regulatory Commission, October 29, 1998.

NRC Licensee Event Report LER 50-249/99-003-01, *Supplement to Reactor Recirculation B Loop, High Pressure Flow Element Venturi Instrument Line Steam Leakage Results in Unit 3 Shutdown Due to Fatigue Failure of Socket Welded Pipe Joint*, U.S. Nuclear Regulatory Commission, August 30, 1999.

NUREG-1544, *Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components*, U.S. Nuclear Regulatory Commission, March 1, 1996.