XI.M6 BWR CONTROL ROD DRIVE RETURN LINE NOZZLE

Program Description

This program is a condition monitoring program for boiling water reactor (BWR) control rod drive return line (CRDRL) nozzles that is based on the staff's recommended position in NUREG-0619 for thermal fatigue. This program is also intended to address stress corrosion cracking (SCC) discussed in NRC IN 2004-08. The augmented inspections performed in accordance with the recommendations in NUREG-0619 supplement those in-service inspections that are required for these nozzles in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, Table IWB-2500-1, as mandated through reference in 10 CFR 50.55a. Thus, this program includes (a) mandatory in-service inspection (ISI) in accordance with the ASME Code, Section XI, Table IWB 2500-1 (2004 edition[5]), and (b) augmented ISI examinations in accordance with applicant's commitments to U.S. Nuclear Regulatory Commission (NRC) Generic Letter (GL) 80-095 to implement the recommendations in NUREG-0619.

Evaluation and Technical Basis

1. ***Scope of Program:*** The program manages the effects of cracking on the intended pressure boundary function of CRDRL nozzles. The scope of this program is applicable to BWRs whose reactor vessel (RV) design includes a welded CRDRL nozzle design. The scope of the program includes CRDRL nozzles and their nozzle-to-RV welds, which are ASME Code Class 1 components. The scope of the program also includes a CRDRL nozzle cap (including any CRDRL nozzle-to-cap welds) if, to mitigate cracking, an applicant has cut the piping to the CRDRL nozzle, and capped the CRDRL nozzle.

2. ***Preventive Actions:*** Activities for preventing or mitigating cracking in CRDRL nozzles are consistent with a BWR facility's past preventive or mitigation actions/activities in its current licensing basis as stated in the applicant's docketed response to NRC GL 80-095 and made to address the recommendations in NUREG-0619. Maintaining high water purity reduces susceptibility to SCC. 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."

3. ***Parameters Monitored/Inspected:*** The aging management program (AMP) manages the effects of cracking on the intended function of the RV, the CRDRL nozzle, and for capped nozzles, the nozzle caps, and cap-to-nozzle welds. For liquid penetrant test (PT) examinations that are implemented in accordance with this AMP, the AMP monitors for linear indications that may be indicative of surface breaking cracks. For the volumetric ultrasonic test (UT) examinations that are performed in accordance with this AMP, the AMP monitors and evaluates signals that may indicate the presence of a planar flaw (crack).

4. ***Detection of Aging Effects:*** The extent and schedule of inspection, as delineated in NUREG-0619, assures detection of cracks before the loss of intended function of the CRDRL nozzles. Inspection and test recommendations include PT of CRDRL nozzle bend radius and bore regions and the RV wall area beneath the nozzle, control rod drive system performance testing, and for capped nozzles, the nozzle caps and cap-to-nozzle welds. The

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

inspection is to include base metal to a distance of one-pipe-wall thickness or 0.5 inches, whichever is greater, on both sides of the weld.

5. ***Monitoring and Trending:*** The inspection schedule of NUREG-0619 provides timely detection of cracks. Indications of cracking are evaluated and trended in accordance with the ASME Code, Section XI, IWB-3100, against applicable acceptance standard criteria that are specified in the ASME Code, Section XI, IWB-3400 or IWB-3500.

6. ***Acceptance Criteria:*** Any cracking is evaluated in accordance with ASME Code, Section XI, IWB-3100 by comparing inspection results with the acceptance standards of ASME Code, Section XI, IWB-3400 and ASME Code, Section XI, IWB-3500.

7. ***Corrective Actions*** Corrective action is performed in conformance with ASME Code, Section XI, 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:*** Cracking of CRDRL nozzle-to-vessel and nozzle-to-cap welds has occurred in several BWR plants (NUREG-0619 and Information Notice 2004-08). The present AMP has been implemented for nearly 30 years and has been found to be effective in managing the effects of cracking on the intended function of CRDRL nozzles.

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.

Letter from D. G. Eisenhut, U.S. Nuclear Regulatory Commission, to R. Gridley, General Electric Company, *forwarding NRC Generic Technical Activity A-10*, January 28, 1980.

NRC Generic Letter 80-095, (Untitled), November 13, 1980.[6]

[6] This GL forwarded NUREG-0619 to members of the U.S nuclear power industry and requested that licensees owning BWR model reactors provide confirmation of their intent to implement the recommendations of NUREG-0619, as applied to the design of their BWRs.

NRC Generic Letter 81-11, (Untitled), February 29, 1981.[7]

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

NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*, U.S. Nuclear Regulatory Commission, November 1980.

[7] This GL was issued primarily to provide additional clarification on the contents of the confirmatory response that was requested in NRC GL 80-095.