As of: 3/3/16 4:28 PM
Received: February 26, 2016
Status: Pending_Post
Tracking No. 1k0-8o6j-xp5w
Comments Due: February 29, 2016
Submission Type: Web

PUBLIC SUBMISSION

Docket: NRC-2015-0251
Draft Guidance Documents for Subsequent License Renewal

Comment On: NRC-2015-0251-0004
Guidance: Subsequent License Renewal

Document: NRC-2015-0251-DRAFT-0011
Comment on FR Doc # 2015-32368

Submitter Information

Name: Sherry Bernhoft
Organization: EPRI

General Comment

12/23/2015
80 FR 79956

Please see the attached comments provided on the behalf of EPRI.

Attachments

9

Letter - Comments on NRC Commission document on GALL

Attachment -EPRIs Comments on the US NRC Generic Aging Lessons Learned (00000002)

EPRI | ELECTRIC POWER RESEARCH INSTITUTE

February 29, 2016

Ms. Cindy Bladey
U.S. Nuclear Regulatory Commission, Office of Administration
Mail Stop: OWFN-12-H08
Washington, DC 20555-0001

Subject: Docket ID NRC-2015-0251
Electric Power Research Institute Comments on U.S. Nuclear Regulatory Commission's Draft Document on Generic Aging Lessons Learned (GALL) For Subsequent License Renewal

Dear Ms. Bladey:

The Electric Power Research Institute, Inc. (EPRI) respectfully submits the enclosed comments on the U.S. Nuclear Regulatory Commission's draft document on Generic Aging Lessons Learned (GALL) for Subsequent License Renewal. EPRI thanks the NRC for the opportunity to comment on this draft guidance.

EPRI has been a leader in the development of technical bases for aging management of critical components since its inception. It is noted that EPRI technical reports are integrated into many of the GALL aging management programs as either general references or with an NRC safety evaluation and approval. The EPRI technical reports to support aging management are living documents, and updated as needed based on research results, worldwide operating experience, and inspection results.

We hope these comments and technical feedback on the draft document will be valuable to the NRC staff in the development of the final GALL report for subsequent license renewal. If you have any questions or need additional information please contact Sherry Bernhoft at 704-595-2740, or sbernhoft@epri.com.

Sincerely,

Tina M. Taylor

Tina Taylor
Director, Strategic Programs

RT
Attachment

Attachment – EPRI's Comments on the US NRC Generic Aging Lessons Learned Report for Subsequent License Renewal

EPRI Comments

The Electric Power Research Institute (EPRI) provides research and development (R&D) leadership and coordination in support of long term operations. The R&D focus includes addressing the aging degradation of systems, structures and components (SCCs) in operating nuclear power plant environments. For aging degradation and aging management, the EPRI research results include identification of degradation mechanisms, understanding their causes, assessment of degradation rates, development of options to mitigate, and the tools and processes to support SCC repairs or replacements as needed.

Research programs to provide the understanding and management of SCC aging have been in place for several decades. EPRI programs, such as the BWR Vessel and Internals Project (BWRVIP), Materials Reliability Program (MRP), and Steam Generator Management Program (SGMP), have generated a number of technical reports that provide the basis for the existing aging management programs (AMPs) currently referenced in the GALL. The EPRI issue management programs are living programs and the technical reports are updated based on research results, worldwide operating experience and inspection results. This process will continue as the nuclear power plants consider Subsequent License Renewal (SLR) and extending to 80 years of operation.

EPRI has supported a number of technical discussions with the NRC staff regarding the research results, status of projects and plans for future research. As the NRC staff completes the work on the GALL for SLR we will continue to support the technical exchanges of research information and operating experience.

Provided below are specific comments for your consideration:

XI.M7, BWR Stress Corrosion Cracking

Elements 3 and 4 of the draft AMP state that examination and inspection methods are delineated in EPRI technical report, BWRVIP-75A *BWR Vessel and Internals Project, Technical Basis for Revision to Generic Letter 88-01 Inspection Schedules* (1012621). BWRVIP-75A does not include guidance relative to examination, inspection methods, or test techniques. It only provides alternative guidance for extent and schedule.

Comment: Reference to BWRVIP-75A relative to examination methods or test techniques should be removed. Retain the references to NUREG-0313, Rev.2, and NRC GL 88-01.

XI.M9, BWR Vessel Internals

Screening of Cast Austenitic Stainless Steel (CASS) Reactor Internals

Page XI.M9-1, rows 19 thru 44 (Program Scope) describes screening criteria applicable to CASS reactor internals and page XI.M9-3, row 28 includes a fluence threshold of $1E^{17}$ n/cm^2 for consideration of fracture toughness in CASS reactor internals.

Comment: There are ongoing activities related to NRC review of the BWRVIP approach for management of CASS internals. The BWRVIP recently received a draft Safety Evaluation (SE) from the NRC Division of Engineering based on their review of BWRVIP-234 *BWR Vessel Internals Project, Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals* (1019060). The review of the draft SE may result in a change to the screening threshold and inspection scope which should be reflected in the final AMP recommendation.

An allowance for an alternative screening criteria is mentioned on page XI.M9.

Comment: The text should be clarified to confirm that one acceptable alternative is the screening criteria associated with BWRVIP-234 and the associated NRC SE.

Top Guide Aging Management

The GALL-SLR states top guide inspection requirements as follows:

"BWRVIP-26-A and BWRVIP-183 provide guidelines for inspection and evaluation; BWRVIP-50-A provides guidelines for repair design criteria. The program inspects 5 percent of the top guide locations using enhanced visual inspection technique, EVT-1 within 6 years after entering the subsequent period of extended operation. An additional 5 percent of the top guide locations will be inspected within 12 years after entering the subsequent period of extended operation."

Comment: For the SLR period, only the reinspection requirements of BWRVIP-183, *BWR Vessel and Internals Project, Top Guide Grid Beam Inspection and Flaw Evaluation Guidance* (1013401), will apply: to inspect 10 percent of grid cells every 12 years. The recommendation to inspect an additional 5 percent of grid cells within 12 years of entering the SLR period is not based on research or operating experience. Also note that inspections may be performed using either EVT-1 or UT (not just EVT-1 as stated in XI.M9). Additionally, for the Top Guide Grid Beams, the NRC recently issued a SE approving the BWRVIP approach described in BWRVIP-183. This should be reflected in the GALL-SLR.

XI.M11B Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components (PWRs only)

NRC has proposed two new recommendations in Section 4-Detection of aging effects of this AMP:

1. Branch Connections: "The program also performs a baseline volumetric or inner-diameter surface inspection of all susceptible nickel alloy branch line connections and associated welds as identified in Table 4-1 of MRP-126 if such components or welds are of a sufficient size to create a loss of coolant accident (LOCA) through a complete failure (guillotine break) or ejection of the component. The baseline inspection is performed prior to the subsequent period of extended operation using a qualified method in accordance with Appendix IV or VIII of ASME Code Section XI as incorporated by reference in 10 CFR 50.55a, or equivalent. Existing periodic inspections using volumetric or surface examination methods may be credited for the baseline inspection. If the baseline inspection indicates the occurrence of PWSCC, periodic volumetric or inner-diameter surface inspections are performed with adequate periodicity."

Comment: MRP-126, *Materials Reliability Program Generic Guidance for Alloy 600 Management* (1009561), documents locations with dissimilar metal welds that could be susceptible to Stress Corrosion Cracking (SCC) in a PWR environment. MRP-139, *Primary System Piping Butt Welds Inspection and Evaluation Guidelines* (1015009), documents dissimilar metal weld locations where

inspection is needed. MRP-139 became the basis for the ASME Code Case, N-770. Based on research and operating experience Code Case N-770 provides reasonable assurance of safety for inspection and evaluation of dissimilar metals welds susceptible to SCC.

2. Bottom Mounted Nozzles (BMN): "In addition, this program performs a baseline inspection of bottom-mounted instrumentation (BMI) nozzles of reactor pressure vessels (RPVs) using a qualified volumetric examination method. The inspection is conducted on all susceptible nickel alloy BMI nozzles prior to the subsequent period of extended operation. If this inspection indicates the occurrence of PWSCC, periodic volumetric inspections are performed on these nozzles and adequate inspection periodicity is established. Alternatively, applicant-proposed and staff-approved mitigation methods may be used to manage the aging effect for these components."

Comment: The MRP's BMN safety assessment, shared with NRC in public meetings, concludes that a program of regular visual examinations provides reasonable assurance for detection of leakage.

XI.M16, A PWR Vessel Internals

DELETED in GALL. To be replaced by plant-specific program.

Comment: Chapter XI.M Aging Management chapters; Consideration should be given to retaining the AMP XI.M16A regarding PWR Reactor Internals aging management. The existing MRP-227-A, *Materials Reliability Program: Pressurized Water Reactor Internals and Evaluation Guidelines* (1022863), for 40-60 years is a technically robust aging management program based on proactive inspections and activities to monitor aging mechanisms in the plants. The techniques and inspection strategies have been proven over many years both within the BWRVIP and PWR-MRP programs. Industry has an on-going initiative through EPRI to establish any supplemental requirements to MRP-227-A that may be needed to address the SLR application for 60-80 years life. This effort is expected to be completed in the 2019-2020 time-frame.

XI.M31, Reactor Vessel Material Surveillance

Page XI.M31-1: "The surveillance program must comply with ASTM International (formerly American Society for Testing and Materials) Standard Practice E 185-82, as incorporated by reference in 10 CFR Part 50, Appendix H."

Comment: GALL-SLR references E185 (as per Appendix H guidelines), Appendix H is being revised to recognize E2215 and E185.

Comment: GALL-SLR should be consistent and reference both ASTM standards.

Page XI.M31-1 Lines 21-24 – States that if standby capsules are going to be included and are not in the vessel, they shall be reinserted. However, on page XI.M31-4 (lines 40-45) it states that if a capsule has already been pulled and has enough fluence it can be tested without inserting it back into the vessel.

Comment: These statements seem to conflict.

Page XI.M31-5, lines 32-37: "If the plant uses an embrittlement trend curve to determine embrittlement (such as those of RG 1.99, Rev. 2, 10 CFR 50.61, and 10 CFR 50.61a), the program ensures that the

operating conditions for the reactor vessel beltline are within the applicability limits of the embrittlement trend curve with respect to parameters such as irradiation temperature, neutron fluence, and flux, or provides technical justification for exceeding these applicability limits."

Comment: This provision modifies the requirements of 10CFR50.61. The guidance needs to be clarified to with regard to intent of the AMP wording.

Page XI.M31-3, lines 11-13: The proposed GALL for SLR states that "This program includes removal and testing of at least one capsule...with a neutron fluence of the capsule between one and one and one quarter (1.25) times the projected peak neutron fluence..."

Comment: This tightens of the target fluence band for the period of SLR. The latest ASTM Standard, E2215, was developed with extended operation being considered, and this standard retains a target of between one and two times end of life fluence.

Irradiation embrittlement is primarily a fluence driven effect. The latest version of ASTM E900 identifies an embrittlement trend curve that has no consideration of flux effects. As such, time effects for surveillance data are of very minor significance and we suggest retaining the prior fluence band.

Electrical Cables

XI.E1, Electrical Insulation for Electrical Cables and Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements

Comment: EPRI Guidance for cable aging management could be used as reference material and be relied upon as a resource for this section. Guidance is provided in the following three reports:

1. For Medium Voltage, *Plant Engineering: Aging Management Program Guidance for Medium-Voltage Cable Systems for Nuclear Power Plants, Revision 1.* (3002000557).
2. For Low Voltage, *Plant Support Engineering: Aging Management Program Development Guidance for AC and DC Low-Voltage Power Cable Systems for Nuclear Power Plants.* (1020804).
3. For I&C, *Plant Support Engineering: Aging Management Program Development Guidance for Instrument and Control Cable Systems for Nuclear Power Plants.* (1021629).

These reports have been made available to the NRC staff for their review under our memorandum of understanding (MOU). In addition there have been a number of technical exchange meetings on cable aging management. As the NRC staff develops the final AMPs EPRI would like to work with the NRC staff to ensure a clear understanding of the technical content and references to EPRI reports for aging management.

XI.E2, Electrical Insulation for Electrical Cables and Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits

Comment: Identification of insulation deterioration by insulation resistance should be replaced with more general terminology such as "cable insulation deterioration shall be monitored by some combination of physio-chemical and mechanical or electrical testing determined by the licensee to be appropriate for the aging conditions identified". This AMP also identifies reduced electrical insulation resistance as the main

way of evaluating cables that will more likely be detected by other physical, mechanical or electrical testing other than insulation resistance.

XI.E3A, Electrical Insulation for Inaccessible Medium Voltage Power Cables Not Subject To 10 CFR 50.49 Environmental Qualification Requirements

Comment: Revise to allow licensee to describe their test methodology to detect degraded/wet cable insulation. This AMP under "Parameters of Aging" and "Detection of Aging" also identifies reduced electrical insulation resistance as the main way of evaluating cables.

Comment: Insulation resistance is not the parameter of concern. Loss of dielectric strength (ac breakdown, high dielectric loss) is the major concern. EPRI's work on "Tan Delta" testing may be a valuable reference. [*Plant Engineering: Evaluation and Insights from Nuclear Power Plant Tan Delta Testing and Data Analysis – Update* (3002005321).]

XI.E3B, Electrical Insulation for Inaccessible Instrument and Control Cables Not Subject To 10 CFR 50.49 Environmental Qualification Requirements

Comment: Make this conditional based on OE. Although there are instances of I&C cables failing in wet locations, no widespread, statistically significant, or research/forensic evidence that a common cause deterioration has been identified. A useful reference is: *Aging Management Program Development Guidance for Instrument and Control Cable Systems for Nuclear Power Plants*. (1021629).

XI.E3C, Electrical Insulation for Inaccessible Low Voltage Power Cables Not Subject To 10 CFR 50.49 Environmental Qualification Requirements

Comment: This requirement could be conditional based on operating experience. Although there are instances of I&C cables failing in wet locations, there is no widespread, statistically significant, or research/forensic evidence that a common cause deterioration has been identified. A useful reference is: *Aging Management Program Development Guidance for AC and DC Low-Voltage Power Cable Systems for Nuclear Power Plants*. (1020804).