October 11, 2012

Mr. Neil Wilmshurst
Vice President & Chief Nuclear Officer
Electric Power Research Institute
1300 West WT Harris Blvd.
Charlotte, NC 28262

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RELATED TO WCAP-17096-NP, REVISION 2, "REACTOR INTERNALS ACCEPTANCE CRITERIA METHODOLOGY AND DATA REQUIREMENTS," DECEMBER 2009 (TAC NO. ME4200)

Dr. Mr. Wilmshurst:

By letter dated May 19, 2010, the Electric Power Research Institute (EPRI), in cooperation with the Pressurized Water Reactor Owner's Group, submitted topical report WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," dated December 2009, for review by the U.S. Nuclear Regulatory Commission (NRC) staff (Agencywide Documents Access and Management System (ADAMS) Accession No. ML101460154). By letter dated June 14, 2012, EPRI submitted a revised response to an NRC staff request for additional information (RAI) (ADAMS Accession No. ML12171A374).

Upon review of the information provided, the NRC staff has determined that additional information is needed to complete the review as detailed in the enclosure. We request that you submit your response to the RAI questions in the enclosure to the NRC Document Control Desk within 20 days of receipt of this letter.

If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-1002.

Sincerely,

/RA/

Joseph A. Golla, Project Manager
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 669

Enclosure:
RAI questions

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REQUEST FOR ADDITIONAL INFORMATION RELATED TO WCAP- 17096-NP, REVISION 2, "REACTOR INTERNALS ACCEPTANCE CRITERIA METHODOLOGY AND DATA REQUIREMENTS"
(TAC NO. ME4200)
PROJECT 669

RAI A

In response to Request for Additional Information (RAI) 30, Part 1a, the Electric Power Research Institute (EPRI) referenced the boiling water reactor (BWR) hydrogen water chemistry (HWC) crack growth rate (CGR) from Carter, Robert and Pathania, Raj, "Technical Basis for BWRVIP [BWR Vessel and Internals Project] Stainless Steel Crack Growth Correlations in BWRs," PVP2007-26618, ASME [American Society of Mechanical Engineers] Pressure Vessel & Piping Conference Proceedings, July 2007 as the basis for stress corrosion cracking CGR in a component with a neutron fluence less than or equal to $5x10^{20}$ neutrons per square centimeter (n/cm^2) at E>1.0 Megaelectronvolts (MeV). The stress intensity factor dependent CGR provided in this reference for HWC conditions is:

> $da/dt = (0.983x10^{-8})*K^{2.181}$ inches per hour (in/hr)[1] for fluence less than or equal to $5x10^{20}$ n/cm^2 at E>1.0 MeV

In its response to RAI 34, part 1b, EPRI additionally referenced the HWC CGR from paragraph C-8520 of Appendix C of Section XI of the 2010 ASME Boiler & Pressure Vessel Code (ASME Code) for the CGR at fluences less than or equal to $5x10^{20}$ n/cm^2 at E>1.0 MeV:

> $da/dt = (5.31x10^{-9})*K^{2.181}$ in/hr for fluence less than or equal to $5x10^{20}$ n/cm^2 at E>1.0 MeV

Additionally, for neutron fluences between $5x10^{20}$ n/cm^2 and $3x10^{21}$ n/cm^2, references are provided for the CGR model in the responses to RAI 30, Part 1a and 34, Part 1b. The RAI responses did not indicate that EPRI plans to include the CGR or the references for the CGR for either fluence range in the final version of WCAP-17096-NP.

The staff therefore requests the following:

1. For neutron fluence $<5x10^{20}$ n/cm^2, since the ASME Code CGR is different than the CGR provided in the referenced paper, the staff requests EPRI clarify in your response to this RAI, which CGR is to be used. Since the ASME CGR is smaller by approximately a factor of 2 than that justified in the referenced paper, if the ASME CGR is used the staff requests EPRI provide additional justification for the lower CGR proposed for neutron fluence less than $5x10^{20}$ n/cm^2.

2. Confirm that WCAP-17096-NP will be modified to specify the use of the appropriate BWR HWC CGR (as modified if necessary by the response to Part 1 of this RAI) as a function of neutron fluence either in the analysis procedures for the subject components, or in a general section of the report, as appropriate.

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[1] where a is the crack dimension in inches, t is time in seconds, and K is the stress intensity in kilopounds per square inch times the square root of inches (ksi√in)

RAI B

1. In its response to RAI 31, Part 1, EPRI provided the following reference for the fracture toughness (K_{IC}) values of 150 ksi√in to be used for evaluations of materials in low neutron fluences regions:

   J.K. McKinley, et al., "CGR and Fracture Toughness of Austenitic Stainless steel in a PWR Primary Water Environment," 14th International Conference on Environmental Degradation of Materials in Nuclear Power Systems, August 23-27, 2009

   However, based on examining Figure 4 of the reference, 150 ksi√in was determined from the best fit line through the data and does not represent a lower bound to the data in the fluence range of 0.5 displacements per atom (dpa) ($3x10^{20}$ n/cm^2) to 1.5 dpa ($1x10^{21}$ n/cm^2).

   The staff requests EPRI justify using a fracture toughness value that is not determined from a lower bound line to the data.

2. Define what is considered a "low fluence" region, in terms of a neutron fluence or dpa value in the analysis procedures for the subject items. EPRI shall also confirm that WCAP-17096-NP will include this definition in its final Revision 2 version in the appropriate section of the report.

RAI C

In RAI 29, Part 5, RAI 31, Part 1, and RAI 34, Parts 2a, 2b, and 2c, EPRI referred to fracture toughness values and recommended evaluation methodologies (as a function of neutron fluence) that are recommended in Section 6.0 of "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" (MRP-227), Rev. 0. In the responses to RAI 31, Part 1 and RAI 34, Part 2a, EPRI cited J.K. McKinley, et al., "CGR and Fracture Toughness of Austenitic Stainless Steel in a PWR [pressurized water reactor] Primary Water Environment," 14th International Conference on Environmental Degradation of Materials in Nuclear Power Systems, August 23-27, 2009, as the basis for the recommended fracture toughness values.

The staff requests that EPRI add the fracture toughness values (as modified if necessary by the response to RAI B) and recommended evaluation methodologies to the methodology sections of the components for which these fracture toughness values and evaluation methodologies are recommended, or add this to a general section of the report as appropriate. Both a response to this RAI and a commitment to update WCAP-17096-NP Revision 2 are requested.

RAI D

In the responses to RAI 29, Part 7, RAI 35, Part 3, and RAI 39, Part 3, EPRI indicated that in the case of multiple cracks, the guidance for the evaluation of allowable crack proximity from the 2004 ASME Code, Section XI will be used.

The staff requests that EPRI confirm that a description of this guidance and a reference to the relevant ASME Code, Section XI guidance will be added to the methodology section for the subject items, or to a general section of the report as appropriate.

RAI E

The response to RAI 33, Part 1, stated in part that testing of irradiated bolts have shown they retain sufficient ductility such that the ASME allowable stresses can still be used, and cited testing of bolts removed from Farley that demonstrated 25 percent strain to failure and reduction in area of 62 percent. The response references internal Westinghouse documents that have not been submitted to the NRC as the source of this information. The staff reviewed information on the tensile properties of irradiated baffle-former bolts reported in “Materials Reliability Program Hot Cell Testing of Baffle/Former Bolts Removed from Two Lead PWR Plants” (MRP-51) 1003069 Final Report, November 2001. With respect to the ductility of the bolting materials MRP-51 concluded that the ductility of the bolting materials remains relatively high with the total elongation to fracture >14 percent at room temperature and >7 percent at 608 °F (320 °C). The tests of the bolts from Farley documented in MRP-51 showed total elongation of 14-19 percent at room temperature and 8.5 percent at 608 °F.

The staff notes that the maximum neutron fluence received by the bolts as reported by MRP-51 was 15 dpa or approximately $1x10^{22}$ n/cm^2, while MRP-191 provides a screening value for neutron fluence (representative of end-of-life or 60 years operation) for baffle-former bolts of $5x10^{22}$ $n/cm.^2$ The response to RAI 33, Part 1, did not provide the fluence value associated with the reported test results.

Based on the above, the staff requests the following:

1. In a response to this RAI, provide the fluence value for the Farley bolts for which the tensile test results were discussed in the response to RAI 33, Part 1.
2. Considering the end-of-life fluence value in MRP-191, in a response to this RAI, discuss whether the Farley bolt tensile testing results can be considered representative of end-of-life conditions, considering the fluence received by the bolts.

RAI F

1. In response to RAI 26, Part 4, EPRI stated “The CRGT [control rod guide tube] lower flange weld inspection strategy is to inspect a sample of welds that can be visually examined without removing the CRGTs. Employing this strategy may result in the examination of less than 100% of an individual CRGT welds. If no degradation is detected, the sample is complete and acceptable. If degradation is found, the plant's corrective action program is entered to determine the path forward by evaluation. This may include expanding the sample, depending on the plant-specific loading conditions. If degradation is detected, evaluation of the specific CRGT(s) must be performed. Justification is required to provide reasonable assurance that the uninspected welds are intact or that their degradation would not affect performance.” EPRI further stated at the end of the RAI 26 response that there is no intention to assume that an uninspected weld is failed.

   Considering the above, clarify in a response to this RAI, how reasonable assurance of the structural integrity of the uninspected CRGT lower flange welds will be achieved regardless of whether degradation is found in the inspected sample. In addition, EPRI shall also confirm that WCAP-17096-NP will include this information in its final Revision 2 version in the appropriate section of the report.

2. In the response to RAI 26, Part 5, EPRI stated that CRGTs returned to service with any flawed welds would be reinspected after one refueling cycle, with the subsequent

reinspection frequency to be determined based on the results of the first subsequent inspection. However, the observation of failed welds in a small number of CRGT lower flanges could represent the leading edge of a time-dependent failure distribution.

The staff requests EPRI discuss in a response to this RAI, the need to reinspect the entire accessible population of CRGT welds, or some smaller sample size larger than the single defective CRGT, if degradation of some welds is found in the initial inspection. In addition, EPRI shall confirm that WCAP-17096-NP will be modified accordingly.

RAI G

In response to RAI 44, Part 1, EPRI stated that if the dynamic response of the thermal shield is changed by removal of one flexure, then resulting loads and stresses in the remaining flexures, bolts, and pins must be evaluated to determine the expected time to failure. This will determine the required interval for reinspection.

The staff requests that EPRI add this guidance to the methodology section for W-ID:8.

RAI H

In response to some of the RAI questions, EPRI indicated that guidance on fracture mechanics methodology from MRP-227, Rev. 0 should be used. (The staff notes that this guidance did not change from MRP-227, Rev. 0 to MRP-227-A). However, EPRI also indicated the fracture mechanics methodology would be modified to be consistent with BWRVIP-76-A. The staff notes there are some differences in the recommended methodologies as a function of fluence in MRP-227-A versus BWRVIP-76-A. MRP-227-A recommends the elastic plastic fracture mechanics (EPFM) method for fluences between 0.5 and 5 dpa, while for a similar fluence range, BWRVIP-76-A recommends the more limiting of linear elastic fracture mechanics (LEFM) or limit load analysis, but allows EPFM as an alternative. MRP-227-A recommends LEFM for fluences 5 dpa ($3x10^{21}$ n/cm^2) and above, with different fracture toughness values for 5-15 dpa and 15 dpa and above. BWRVIP-76-A recommends LEFM for fluence equal to or greater than $3x10^{21}$ n/cm^2.

The staff requests EPRI clarify in a response to this RAI, which set of recommendations for fracture mechanics methods will be used, those of MRP-227-A, Section 6.0 or those of BWRVIP-76-A. In addition, EPRI shall confirm that WCAP-17096-NP will be modified accordingly.

NRC Technical Contact: Jeffrey Poehler

NRC Project Manager: Sheldon Stuchell