August 18, 2014

B. C. Rudell
Chairman, Integration Committee
EPRI-Materials Reliability Program
3420 Hillview Avenue
Palo Alto, CA 94304-1395

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

Dear Mr. Rudell:

By letter dated May 19, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML101460156), the Electric Power Research Institute (EPRI) in cooperation with the Pressurized Water Reactor Owners 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. By letters dated June 14, 2012, and January 3, 2013 (ADAMS Accession Nos. ML12171A374 and ML13008A161), EPRI submitted responses to NRC staff requests for additional information (RAIs). By letters dated April 10, 2014 (ADAMS Accession No. ML14104B579) and July 8, 2014 (ADAMS Accession No. ML14191A014), EPRI submitted proposed changes to the evaluation procedures of WCAP-17096-NP incorporating changes resulting from request for additional information responses, which also includes procedures for additional “Primary” and “Expansion” category components added due to conditions of the staff’s final safety evaluation of MRP-227.

Upon review of the information provided, the NRC staff has determined that additional information is needed to complete the review. The additional information needed is detailed in the enclosure.

In an email dated June 25, 2014, Mr. Kyle Amberge, representing EPRI, agreed that the NRC staff will receive your response to the enclosed questions, approximately September 30, 2014. If you have any questions regarding the enclosed questions, please contact me at 301-415-7297.

Sincerely,

/RA/

Joseph J. Holonich, Senior Project Manager
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 669

Enclosure:
As Stated

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ADAMS Accession No.: ML14177A071 **NRR-088**

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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)

Background

The Electric Power Research Institute Material Reliability Program letter dated April 10, 2014 (Agencywide Documents Access and Management System Accession No. ML14104B579), and July 8, 2014 (ADAMS Accession No. ML14191A014) contained proposed revised procedures for a number of reactor vessel internals components. Proposed evaluation procedures have been received for the following welds for which MRP-227-A specifies or allows visual examinations to be conducted from one side of the weld only:

- CE-ID:6 Core Support Barrel Assembly – Upper (Core Support Barrel) Flange Weld
- CE-ID:6.1 Core Support Barrel Assembly – Lower Core Barrel Flange
- CE-ID:6.2 Core Support Barrel Assembly – Upper Cylinder (Including Welds)
- CE-ID:7, Core Support Barrel Assembly – Lower Cylinder Girth Welds
- CE-ID:7.1 Core Support Barrel Assembly – Core Barrel Assembly Axial Welds
- W-ID:3 Core Barrel Assembly – Upper Core Barrel Flange Weld
- W-ID:4 Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Girth Welds
- W-ID:4.1 Core Barrel Assembly – Upper and Lower Core Barrel Cylinder Axial Welds
- W-ID:5 Core Barrel Assembly – Lower Core Barrel Flange Weld

The following discussion applies to all the welds listed above, for which the proposed evaluation procedures are virtually identical with respect to the assumptions made in determining crack growth. Under "Inputs and Assumptions," the procedures include the following:

"The inspections identified in MRP-227-A are intended to provide a sampling of potential locations of degradation. Under this approach, inspection of one side (surface) of the weld is assumed to provide an adequate sampling for monitoring stress corrosion cracking (SCC)."

Under "Data Requirements," the procedures, with respect to "Flaw Depth", state (in part) that (a) For one-sided visual inspections, the flaw is assumed to be through-wall, and (b) supplemental examinations may be used to determine flaw depth for a flaw-specific criterion, if needed.

Also under "Data Requirement," Item 4 states (in part) (b) Stresses which have an insignificant net-through-wall value (average stress is near zero), such as weld residual stresses and thermal stresses due to local through-wall temperature gradients are considered to have minimal impact on the effective crack growth rates in through-wall flaws, and (c) secondary weld residual and thermal stresses need to be considered in determination of axial and through-wall crack growth rates in partial through-wall flaws, whose dimensions would have to be determined with supplemental ultrasonic testing examinations.

ENCLOSURE

Under analysis, the procedures state, in part, that:

"All analyses require an assumption of the SCC/irradiation-assisted stress corrosion cracking (IASCC) crack growth expected over the upcoming period of service. The methodology is based on analysis of a through-wall flaw with weld residual and thermal stresses relieved."

<u>Request</u>

1. Since supplementary inspections to determine the crack depth are allowed but not required by the evaluation procedure, a visually observed crack may not be throughwall. Considering that weld residual and thermal stresses would still act on a non-throughwall crack, to support the assumption that all visually observed cracks are throughwall is conservative, demonstrate that a non-throughwall crack will not grow at higher rate such that it would attain critical size prior to the next scheduled inspection. This demonstration may be done generically, or the evaluation procedures may be modified to require such a demonstration as part of the evaluation of the specific weld in which degradation has been detected.

2. Under "Analysis," the procedures state that in order to apply the acceptance criteria to a full 10-year inspection interval, follow-up action is required to verify the assumptions used in the predicted crack-growth rate, and that a re-inspection of the indication at a future specified outage, for example, would provide data that could be used to satisfy this verification requirement. Modify the evaluation procedures to address the following:

    a. Define the re-inspection time, e.g. the first refueling outage after the initial inspection, and describe how the required re-inspection time is determined.
    b. Provide details about any follow-up action other than re-inspection, and justify how such actions could verify the predicted crack growth rate.