October 20, 2009

Mr. Christian B. Larsen
Nuclear Vice President & Chief Officer
Electric Power Research Institute
3420 Hillview Avenue
Palo Alto, CA 94304-1338

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1018427, "NONDESTRUCTIVE EVALUATION: PROBABILISTIC RISK ASSESSMENT TECHNICAL ADEQUACY GUIDANCE FOR RISK-INFORMED IN-SERVICE INSPECTION PROGRAMS" (TAC NO. ME1057)

Dear Mr. Larsen:

By letter dated February 18, 2009, Electric Power Research Institute (EPRI) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report 1018427, "Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs." Upon review of the information provided, the NRC staff has determined that additional information is needed to complete the review. On September 21, 2009, Patrick O'Regan, EPRI Project Manager, and I agreed that the NRC staff will receive your response to the enclosed Request for Additional Information (RAI) within 60 days of issuance of this letter. During a conference call with EPRI on September 24, 2009, the NRC staff discussed the draft RAI with the EPRI representatives. At the conclusion of the call, the NRC staff recognized that RAI-6c was asked in error and have agreed per this letter to withdraw RAI-6c. If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-3610.

Project No. 669

Enclosure: RAI questions

cc w/encl: See next page

Sincerely,

/RA/

Tanya M. Mensah, Senior Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

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REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT 1018427, "NONDESTRUCTIVE EVALUATION: PROBABILISTIC RISK ASSESSMENT TECHNICAL ADEQUACY GUIDANCE FOR RISK-INFORMED IN-SERVICE INSPECTION PROGRAMS"

ELECTRIC POWER RESEARCH INSTITUTE

PROJECT NO. 669

The NRC staff has reviewed the Electric Power Research Institute (EPRI) Topical Report (TR) 1018427, "Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs." Based on the review of EPRI TR 1018427, the NRC staff is requesting additional information to complete the review. EPRI TR 1018427 references the American Society of Mechanical Engineers (ASME), Probabilistic Risk Assessment (PRA) Standard (ASME RA-Sb-2005)[1] that was prepared in 2005 as endorsed by Regulatory Guide (RG) 1.200[2], Revision 1 in 2007, with respect to PRA technical adequacy.

RAI-1 The EPRI TR 1018427 fails to provide general guidelines which describe the overarching framework from which acceptable capability categories (CCs) for individual supporting requirement (SRs) for the internal events PRA can be determined. An example of a general guideline that is included is the explanation in EPRI TR 1018427 that SRs that solely address quantitative attributes are of limited importance. The risk ranking and change in risk estimates in EPRI's risk-informed inservice inspection (RI-ISI) methods use an order of magnitude approach which reduces the influence of PRA elements that might only change the quantitative results slightly. However, other general elements such as importance of logic modeling and human actions in the internal events PRA should be likewise generally characterized. For example, it would appear that the internal event PRA logic models need to be of relatively high quality (i.e., accurate and high resolution) because multiple consequential structure, system, and component failures need to be evaluated using these logic models. Please identify general guidelines for the technical elements that compose an internal events Level 1/Large Early Release Frequency (LERF) PRA based on how EPRI's RI-ISI method relies on these elements.

RAI-2 Due to the lack of general guidelines, many of the discussions on individual SRs appear to be conclusions with no justification. Based on the general guidelines developed for RAI-1, please re-evaluate target categories for the specific SRs in the internal events PRA and indentify which general guideline supports the selected category.

RAI-3 The EPRI TR 1018427 only provides guidance in defining the applicable ASME PRA Standard SRs and the appropriate CCs for the Levels 1 and 2 analyses of internal events while

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[1] ASME RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum B to ASME RA-S-2002, ASME, New York, New York, December 30, 2005.

[2] "An Approach for Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities," Revision 1, January 2007.

at power. The EPRI TR 1018427 states that, "As future revisions to RG 1.200 occur, this work will be updated to support future RI-ISI application and maintenance."

a) It is acknowledged that ASME and the American Nuclear Society (ANS) have issued a combined standard "ASME/ANS RA-Sa-2009" in February 2009, and endorsed in RG 1.200, Revision 2, in March 2009. The EPRI should provide its position on this combined standard in support of the RI-ISI PRA technical adequacy including the following hazard groups:

- Internal Fires
- Seismic Events
- High Winds
- External Floods, and
- Other External Hazards

b) Discuss whether the guidance provided in EPRI TR 1018427 would be treated differently for operating plants and plants licensed under 10 CFR Part 52, "Licenses, Certifications, and Approvals For Nuclear Power Plants."

RAI-4 EPRI TR 1018427, Page V, second paragraph "Results and Findings." The second sentence states that, "The technical adequacy of the PRA is determined by demonstrating that the PRA meets technical elements and associated SRs of NRC RG 1.200." It should be noted that RG 1.200 provides the NRC staff's position on supporting requirements (including clarifications as needed), but does not provide supporting requirements as stated in the above statement. Please clarify the above sentence.

RAI-5 EPRI TR 1018427, Page V, last paragraph "EPRI Perspective." This states that "The vast majority of U.S. plants that implement RI-ISI programs have used tools and products developed by the EPRI. The EPRI TR 1018427 reviews these tools and products against the ASME PRA Standard and RG 1.200." Please define the "tools and products" mentioned in this paragraph.

RAI-6 Based on the review of previous RI-ISI submittals that are based, in part, on ASME Code Case N-716[3], the NRC staff believes that additional work may be needed beyond the CCs recommended in EPRI TR 1018427 in order to provide confidence that all high-safety-significant (HSS) segments will be indentified and that an appropriate change in risk is estimated.

a) The EPRI TR 1018427 proposes CC I/II as being sufficient for SR IF-D3a (IFEV-A3). Capability Category I/II permits grouping or subsuming flood-initiated scenarios with existing plant initiating event (IE) groups. Capability category III does not permit subsuming flood IEs with other initiators. A RI-ISI analysis may be done long after the flooding analysis is completed and subsuming flood scenarios into existing plant IE

[3] ASME Code Case N-716, Alternative Piping Classification and Examination Requirements, Section XI Division 1, ASME, New York, New York, April 19, 2006.

groups would require an extra step in the RI-ISI analyses to retrieve any subsumed scenarios. This requirement is mentioned in the table in Appendix A of EPRI TR 1018427I. Please propose changes to the ASME Code Case N-716 methodology that clarifies this additional step or change the recommended CC to Capability category III.

b) The EPRI TR 1018427 proposes CC II as being sufficient for SR IF-C6 (IFSN-A14) and IF-C8 (IFSN-A16). Capability category II permits screening-out of flood areas and sources respectively based on, in part, the success of human actions to isolate and terminate the flood before equipment is damaged. Capability Category III does not permit screening out areas and sources based on reliance on operator actions. Qualitative screening of flood scenarios based on possible human intervention does not appear to be fully consistent with the CCDP/CLERP or CDF/LERF significance determination. The EPRI TR 1018427 simply states that the qualitative screening provides confidence in the high reliability of the human actions. Please explain how the qualitative screening in CC II provides confidence that the quantitative guidelines will not be exceeded or change the recommended CC to Capability category III.

c) The following question was asked in error by the NRC staff and is being withdrawn:

   The EPRI TR 1018427 proposes CC I as being sufficient for SR IF-D5a (IFEV-A6) when determining IE flooding frequencies. Capability category I permits determining flooding IE frequencies using solely generic experience. Capability category II/III requires using a combination of generic and plant-specific experience. RI-ISI is directed toward inspecting locations with the highest risk, driven mostly by failure frequency. Code Case N-716 requires even greater fidelity in the IE frequency because it is identifies HSS segments based on risk. The EPRI TR 1018427 states that CC I is sufficient because that level of detail is sufficient to identify the relative importance of the segments. The staff disagrees. RI-ISI is developed on a plant-specific basis using the absolute importance of segments to risk. Please further explain why plant operating experience reflected in the flooding failure frequencies is not judged to be important in the RI-ISI process or change the recommended CC to Capability category II/III.