July 13, 2009

MEMORANDUM TO: Stacey L. Rosenberg, Chief
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

FROM: Holly D. Cruz, Project Manager **/RA/**
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

SUBJECT: SUMMARY OF THE MAY 13, 2009, CATEGORY 2 PUBLIC MEETING BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND THE PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG) TO DISCUSS ISSUES RELATED TO PWROG TOPICAL REPORT BAW-2374, REVISION 2, "RISK-INFORMED ASSESSMENT OF ONCE-THROUGH STEAM GENERATOR TUBE THERMAL LOADS DUE TO BREAKS IN REACTOR COOLANT SYSTEM UPPER HOT LEG LARGE-BORE PIPING" (TAC NO. MD7404)

On May 13, 2009, a Category 2 public meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and representatives of the Pressurized Water Reactor Owners Group (PWROG) at NRC Headquarters, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to discuss issues related to PWROG Topical Report (TR) BAW-2374, Revision 2, "Risk-informed Assessment of Once-Through Steam Generator Tube Thermal Loads Due to Breaks in Reactor Coolant System Upper Hot Leg Large-Bore Piping." A list of meeting attendees is enclosed and the meeting agenda is contained in the May 13, 2009, public meeting notice available in the Agencywide Documents Access and Management System (ADAMS) as Accession No. ML091210268.

For this TR, the PWROG members attending included both representatives from AREVA NP Inc. (AREVA) and the owners of Babcock & Wilcox (B&W) plants. In summary, the NRC staff met with the PWROG to discuss TR BAW-2374 and the ability of the affected steam generators (SGs) to withstand loads from a postulated design basis candy cane loss-of-coolant accident (LOCA) on a plant-specific basis to gain assurance that there is a common understanding of the analysis requirement to meet 10 CFR 50.46, gather information on whether licensees currently meet 10 CFR 50.46 and to ensure continued compliance in the future as licensee disposition SG tube indications.

Opening remarks were provided by Mr. Tom Blount, Deputy Director, Division of Policy and Rulemaking (DPR), Office of Nuclear Reactor Regulation (NRR). Mr. Blount thanked the members of the PWROG for their efforts to resolve the issues and provided background information and timelines associated with the TR.

Mr. Blount continued with the review of Enclosure 2, and the PWROG provided information on the replacement status of the affected steam generators in its meeting handout (ADAMS Accession No. ML091830557). Of the seven units, four have replaced their steam generators, Oconee Units 1, 2, and 3, and Arkansas Nuclear One (ANO) -1. Two other units, Crystal River 3 and Three Mile Island (TMI)-1, have plans to replace their steam generators in the fall of 2009. Davis-Besse has indicated it plans to replace its steam generators in 2014.

Oconee Units 1, 2, and 3, and the Crystal River 3 (replacement) SGs were designed and manufactured in Canada. The Davis Besse planned replacement SGs will also be designed and manufactured in Canada. The licensees and AREVA indicated that they had confidence that the analyses performed for the design of these SGs would demonstrate that tube integrity could be maintained for the spectrum of design basis LOCAs including one in the candy-cane region of the reactor coolant system (RCS). However, the licensees stated that additional review or analysis would be performed to confirm this.

The ANO-1 and TMI-1 (replacement) SGs were designed and manufactured in France. Although the licensees were confident that the integrity of the SG tubes could be maintained for all design basis LOCAs at these plants, the design and analysis of the SGs may not have explicitly considered the thermal loads from a candy-cane LOCA. AREVA is assessing what additional analysis may be needed for these French-manufactured SGs. Nonetheless, the licensees and AREVA were confident that because the design of the French-made SGs was similar to the Canadian-made SGs, any additional analyses would confirm their expectation that tube integrity will be maintained following all large break LOCAs.

The following next steps were identified as outcomes from the meeting (note: licensees affected by the TR, and noted below attended the meeting as PWROG representatives):

1. The licensees agreed to review their justifications for continued operation that were prepared earlier this decade to ensure that they remained valid.

2. The licensees agreed to reevaluate whether they had implemented sufficient compensatory measures and had appropriately incorporated them into their procedures.

3. The licensees agreed to reassess whether this issue was reportable in accordance with 10 CFR 50.46.

4. The licensees agreed to develop a plan for completing the appropriate additional analyses that would demonstrate tube integrity would be maintained for a large break LOCA in the candy-cane region of the RCS.

5. The licensees agreed to confirm that their tube integrity programs are assessing tube integrity for all large break LOCAs.

Finally, the NRC staff and the PWROG agreed to meet again in June 2009 to discuss these issues including the licensee’s plans for completing the confirmatory SG analyses.

The PWROG handout is available in ADAMS as Accession No. ML091830557.

Members of the public were not in attendance. Public Meeting Feedback forms were not received.

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Project No(s). 694

Enclosures:
1. List of Attendees
2. Background Information for May 13, 2009 Public Meeting

cc w/encls: See next page

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ADAMS Accession No: ML091811290 (Package); ML091210268 (Notice); ML091811289 (Summary) NRC-001

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ENCLOSURE 1

Background Information for May 13, 2009,
Public Meeting with the PWROG regarding TR BAW-2374, Revision 2

Background

The NRC staff considers the topical report request to be a significant departure from the Commission's defense-in-depth philosophy. That is, if the topical report were to be approved, implementing plants would not maintain a multiple-barrier approach against fission product release. The NRC will consider any variations to this philosophy only after considering such factors as the risk and a demonstrated need for the variation.

The NRC staff believes the demonstrated need, described as a "significant" burden in the topical report with respect to meeting the steam generator performance criteria, is plant-specific and should, therefore, be demonstrated separately for each plant. The burden needs to be based upon an actual assessment of tube integrity that includes the loads from postulated design basis candy cane loss-of-coolant accidents (LOCA).

Purpose of Meeting

In order to determine the extent of changes to integrity requirements (defense-in-depth) that may be justifiable, the following information is needed from each licensee:

1. The results of a steam generator tube integrity assessment (including plugs, welds, and repair hardware), when the thermal loads from a postulated design basis candy cane break are included.

2. The burden (hardware, procedural, operational) associated with continuing to meet the steam generator tube integrity requirements, given the steam generator tube integrity assessment results from 1) above. For example, discuss the burden associated with making procedural, operational, or hardware changes that limit the loading on the tubes such that steam generator tube integrity will be assured during a postulated design basis candy cane LOCA.

3. In the event that steam generator tube integrity can not be demonstrated in 1) above, discuss the extent to which compensatory measures can or have been implemented (e.g., procedures for limiting the loss of coolant through the secondary side of the steam generator).

The purpose of this meeting is to discuss, with each licensee, the results of such an assessment or their plans to provide such an assessment.

Note: Although a postulated design basis candy cane LOCA is referenced, it is the intent that the tubes would be able to meet the steam generator tube integrity performance criteria regardless of the location of the LOCA.

ENCLOSURE 2

cc:

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