UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 4, 2009

Vice President, Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 - ONCE-THROUGH STEAM GENERATOR TUBE LOADS UNDER CONDITIONS RESULTING FROM POSTULATED BREAKS IN REACTOR COOLANT SYSTEM UPPER HOT-LEG LARGE-BORE PIPING (TAC NO. MD7178)

Dear Sir or Madam:

On June 25, 2009, a 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, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, regarding once-through steam generator tube loads under conditions resulting from postulated breaks in reactor coolant system upper hot-leg large-bore piping.

In the meeting, the NRC requested that each Babcock and Wilcox licensee submit a letter summarizing the information discussed at the meeting. The NRC staff followed up this verbal request with a letter dated July 31, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092090249).

By letter dated August 31, 2009 (ADAMS Accession No. ML092530659), Entergy Operations, Inc. (the licensee), provided its response for Arkansas Nuclear One, Unit No. 1. In addition, by letter dated October 26, 2009 (ADAMS Accession No. ML093010329), you provided additional information pertaining to the steam generators. The NRC staff has reviewed the response and provides the following comments:

- The staff has noted your response on the reporting requirements of Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," of Title 10 of the *Code of Federal Regulations.* We will contact you if we wish to have further discussions regarding your response.

- Based on its review of these letters, the staff understands that you are verifying that the design of your steam generators is sufficient to withstand large break loss-of-coolant accident (LBLOCA) loading conditions and that steam generator tube integrity is being maintained for all LBLOCAs (including those in the candy-cane region) as required by Technical Specification (TS) 3.4.16, "Steam Generator (SG) Tube Integrity," and TS 5.5.9, "Steam Generator (SG) Program."

- The staff expects that any design/licensing basis documents, including the Updated Final Safety Analyses Report, will be updated, or already has been updated, to reflect your steam generator design and your management of steam generator tube integrity for LBLOCA loads. This includes removing any reference to the approach discussed in the previously withdrawn 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," or similar approach.

If you have any questions, please contact me at (301) 415-1480 or by e-mail at kaly.kalyanam@nrc.gov.

Sincerely,

Kalyanam

N. Kaly Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

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- The staff expects that any design/licensing basis documents, including the Updated Final Safety Analyses Report, will be updated, or already has been updated, to reflect your steam generator design and your management of steam generator tube integrity for LBLOCA loads. This includes removing any reference to the approach discussed in the previously withdrawn 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," or similar approach.

If you have any questions, please contact me at (301) 415-1480 or by e-mail at kaly.kalyanam@nrc.gov.

Sincerely,

/RA/

N. Kaly Kalyanam, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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