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

November 6, 2009

Mr. Barry S. Allen
Site Vice President
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
Mail Stop A-DB-3080
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, 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. ME1798)

Dear Mr. Allen:

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. ML092120411).

In response, FirstEnergy Nuclear Operating Company (the licensee) provided the response in its letter dated August 31, 2009 (ADAMS Accession No. ML092450685), for Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse). 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.

- In its August 31, 2009, letter, the licensee indicated that all flaws were evaluated against worst case large break loss-of-coolant accident (LBLOCA) loading conditions. The results of these analyses indicated that the amount of primary-to-secondary leakage following an LBLOCA was 0.0 gallons per minute (gpm) from steam generator 1B and 0.16 gpm from steam generator 2A. Analyses such as these can be very complex particularly for plants with Alloy 600 tubing material since this material has exhibited a variety of forms of degradation. Another complicating factor may be that the whole tube length was not inspected since Technical Specification 6.8.4.g.4 does not require the portion of the tube outboard of a re-roll to be inspected. As a result of these issues, the staff would like to obtain a better understanding of how these tube integrity analyses were performed for the LBLOCA. Specific issues the staff would like to obtain additional information on are enclosed.

- The staff expects expect 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.

Recognizing that the information requested above could be shared in a number of different ways (e.g., formal meeting, written responses, on-site meeting/audit), the staff is open to suggestions on the best way to share this information.

This closes TAC No. ME1798. If you have any questions, please contact me at 301-415-3719 or by e-mail at Stephen.Sands@nrc.gov.

Sincerely,

C. Goodwin FOR

S. Sands, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosure:
As stated

cc w/encl: Distribution via Listserv

SPECIFIC ISSUES ON TUBE INTEGRITY ANALYSES

PERFORMED FOR THE LARGE BREAK LOSS-OF-COOLANT ACCIDENT

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-346

1. Please discuss how the following issues were addressed in the condition monitoring and operational assessment of tube integrity following a large break loss-of-coolant accident (LBLOCA).

    a. Any flaw with a circumferential component including wear, intergranular attack, and circumferential cracking (including those outboard of the re-rolls) could potentially open up and possibly sever under the axial loads associated with a hot-leg LBLOCA. Please discuss how Davis-Besse Nuclear Power Station, Unit No. 1 addressed the amount of leakage for each of the flaw types with a circumferential component. Please include in the response the structural limit for each of these flaw types for an LBLOCA.

    b. During an LBLOCA, axially oriented flaws that are through-wall and other repair hardware (e.g., plugs, sleeves) could potentially leak. Please discuss how these sources of leakage were accounted for in the analyses.

2. In addition, please provide the following information:

    a. A high-level summary of your analysis methodology.

    b. A summary of the technical basis for your analysis methodology.

    c. The basis for any assumptions (e.g., assumptions on sizing uncertainty).

    d. The primary-to-secondary leakage rate assumed in your accident analysis for a loss-of-coolant accident.

    e. A copy of any reference cited in the above information (if these references are not already publicly available).

since Technical Specification 6.8.4.g.4 does not require the portion of the tube outboard of a re-roll to be inspected. As a result of these issues, the staff would like to obtain a better understanding of how these tube integrity analyses were performed for the LBLOCA. Specific issues the staff would like to obtain additional information on are enclosed.

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/RA by C. Goodwin for S. Sands/

S. Sands, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosure:
As stated

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ADAMS Accession No. ML093010483 **MTM concurred 11/02/09**

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