June 18, 2008

Mr. Barry S. Allen
Site Vice President
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
Mail Stop A-DB-3080
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SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 - REQUEST FOR ADDITIONAL INFORMATION RELATED TO IMPROVED TECHNICAL SPECIFICATIONS CONVERSION (MD6398)

Dear Mr. Allen:

By letter to the Nuclear Regulatory Commission (NRC) dated August 3, 2007, FirstEnergy Nuclear Operating Company (FENOC) submitted a request to an application requesting to amend the operating license, for the Davis-Besse Nuclear Power Station, Unit No. 1. FENOC has proposed to revise the current technical specifications to the improved technical specifications consistent with improved standard technical specifications (STS) as described in "Standard Technical Specifications Babcock and Wilcox Plants," Revision 3.1. STS Revision 3.1 is the December 2005, update to NUREG-1430, which was published June 2004.

The NRC staff is reviewing your submittal and has determined that additional information is required to complete the review. The specific information requested is addressed in the enclosure to this letter. During a discussion with your staff on June 4, 2008, it was agreed that you would provide a response within 30 days from the date of this letter.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. If circumstances result in the need to revise the requested response date, please contact me at (301) 415-4037.

Sincerely,

/RA/

Thomas J. Wengert, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosure:
Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

In reviewing the FirstEnergy Nuclear Operating Company's submittal dated August 3, 2007, related to revising the current technical specifications (CTS) to the improved technical specifications (ITS) consistent with improved standard technical specifications (STS) as described in "Standard Technical Specifications Babcock and Wilcox (B&W) Plants," Revision 3.1, for the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS), the NRC staff has determined that the following information is needed in order to complete its review:

DBNPS Borated Water Storage Tank (BWST)

1. The following reference is made in the DBNPS Updated Final Safety Analysis Report (UFSAR) on page 5.2-2, under the heading 5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY (RCPB):

   "Reactor vessels with lower power ratings but similar geometries and service conditions have been analyzed to demonstrate that the reactor vessel can safely accommodate the rapid temperature change associated with the postulated operation of the Emergency Core Cooling System (ECCS) at the end of the vessel's design life. The evaluation is summarized as follows: The state of stress in the vessel during the LOCA [loss-of-coolant accident] was evaluated for an initial vessel temperature of 608˚F. The inside of the vessel wall is rapidly subjected to 90˚F injection water of the maximum flow rate obtainable. The results show that the integrity of the vessel is not violated."

   The reactor vessel stress analysis discussed in the above UFSAR statement appears to be inconsistent with the 35˚F minimum temperature for the DBNPS BWST (a) in the facility's CTS and (b) which is being proposed for the ITS conversion.

   a. Provide additional information to clarify the analysis which is being referred to in the above UFSAR quote.

   b. Does this refer to a stress analysis performed to demonstrate compliance with American Society of Mechanical Engineers Code stress limits or was this analysis performed for some other purpose?

   c. Has the analysis referenced in the UFSAR been performed with the assumption of 35˚F injection water? Why or why not?

2. The basis provided for the selection of 35˚F as the BWST minimum temperature, in the proposed DBNPS ITS Bases, is:

"The 35 ˚F lower limit on the temperature of the solution in the BWST is assumed for the containment vessel vacuum breaker sizing. This temperature also helps prevent boron precipitation."

However, the basis provided for selection of [40 ˚F] as the BWST minimum temperature, in the B&W Design STS Bases, is:

"The 40 ˚F lower limit on the temperature of the solution in the BWST was established to ensure that the solution will not freeze. This temperature also helps prevent boron precipitation and ensures that water injection in the reactor vessel will not be colder than the lowest temperature assumed in reactor vessel stress analysis."

Conceptually, the statement from the B&W Design STS Bases appears to be referring to an analysis very similar to that discussed on page 5.2-2 of the DBNPS UFSAR. If so,

a. Explain why the analysis referenced in the DBNPS UFSAR does not need to be re-performed using a 35˚F injection water assumption to verify that the basis you suggested for the DBNPS BWST minimum temperature is the bounding consideration for establishing that limit.

Additional Background Information and Regulatory Bases for the Request for Additional Information (RAI)

The license amendment request (LAR) proposes to revise the DBNPS CTS to the ITS consistent with STS as described in NUREG-1430, "Standard Technical Specifications - Babcock and Wilcox Plants" as updated by Revision 3.1 to the STS.

Specifically, the LAR seeks to adopt STS Bases associated with STS Surveillance Requirement (SR) 3.5.4.1 with a number of deviations. Among these deviations proposed in the ITS BASES is the omission of the phrase "ensures that water injection in the reactor vessel will not be colder than the lowest temperature assumed in reactor vessel stress analysis."

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36 requires, in part, that technical specifications (TS) be derived from the analyses and evaluation included in the safety analysis report and that they be accompanied by a summary statement of the bases or reasons for them. The phrase which the LAR seeks to omit was incorporated in the STS Bases in accordance with 10 CFR 50.36 as one of several summary reasons or bases for the minimum allowed BWST temperature limit. 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," 10 CFR 50.55a, "Codes and Standards," Appendix A to Part 50, and "General Design Criteria for Nuclear Power Plants," are incorporated into the STS Bases and into the DBNPS Current Licensing Basis.

Paragraph IV, "The Commission Policy," of 58 FR 39132 "Final Policy on § 50.36 Technical Specifications," clarifies the Commission's expectations regarding the content of the TS Bases. It states, in part:

"Each Limiting Condition for Operation [LCO], Action, and Surveillance Requirement should have supporting Bases. The Bases should at a minimum address the following questions and cite references to appropriate licensing documentation (e.g., FSAR, Topical Report) to support the Bases… What are the Bases for each Surveillance Requirement and Surveillance Frequency; i.e., what specific functional requirement is the surveillance designed to verify? Why is this surveillance necessary at the specified frequency to assure that the system or component function is maintained, that facility operation will be within the Safety Limits, and that the LCO will be met?"

DBNPS Emergency Diesel Generator (EDG)

1. Provide the loading profile for the Appendix R scenario to demonstrate that the proposed EDG endurance/margin test ensures that the analyzed functions can be performed. Alternately, confirm by calculation that the EDG Appendix R loading after 2 hours will be less than 100 percent of its continuous rating, considering the 60.5 hertz maximum steady state frequency proposed by the licensee in the ITS.

Additional Background Information and Regulatory Bases for the RAI

During the review of ITS SR 3.8.1.13, the NRC staff identified an issue which requires additional information. For the purpose of verification of EDG loading values, DBNPS provided an excerpt from the AC Power System Analysis calculation (C-EE-015.03-008, Revision 4). The review of the EDG loading results indicated that EDG 1-1 has a higher loading for the Appendix R scenario than for the loss of off-site power/loss of coolant accident (LOOP/LOCA) scenario. During the Appendix R scenario, the EDG can be loaded up to 2627 kW, which is 101 percent of the continuous rating. The continuous rating of the EDG is 2600 kW.

The purpose of EDG testing at 105 to 110 percent of its continuous rating is to demonstrate that the EDG has adequate margin during the sequencing of various ECCS loads, while the purpose of testing at 90 to 100 percent is to demonstrate long term EDG capability when the loads are expected to be less than the EDG continuous rating. If the analyzed loads are higher than the continuous rating of the EDG, then the actual load profile needs to be followed to ensure that the EDG can perform its analyzed function. The analysis performed on the Appendix R scenario shows that the EDG loading is at 101 percent, which is above the continuous rating of the EDG.

The purpose of SR 3.8.1.13 (endurance/margin run for the EDG) is to demonstrate that EDG can perform its analyzed function. Since the same EDG is used to mitigate both the Appendix R scenario and the LOOP/LOCA scenario, it is essential that the EDGs be tested to the highest loading scenario (worst case scenario) for an adequate duration to demonstrate that EDGs will continue to perform their analyzed function.

The following regulations are considered applicable to testing of the EDGs:

Section 50.36 (d)(2)(ii)(D) of 10 CFR - Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Section 50.36 (d)(2) of 10 CFR - LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility.

Section 50.36 (d)(3) of 10 CFR - SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that the facility operation will be within safety limits, and that the LCO will be met.