July 25, 2007

Mr. Mark B. Bezilla
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 - REQUEST FOR ADDITIONAL INFORMATION RELATED TO MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NO. MD5240)

Dear Mr. Bezilla:

By letter to the Nuclear Regulatory Commission (NRC) dated April 12, 2007, FirstEnergy Nuclear Operating Company submitted a license amendment request for measurement uncertainty recapture power uprate, to 2817 Megawatts thermal (MWt), 1.63 percent above the currently licensed power level of 2772 MWt, for the Davis-Besse Nuclear Power Station, Unit No. 1.

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. Please provide your 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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cc:

Manager, Site Regulatory Compliance
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
Mail Stop A-DB-3065
5501 North State Route 2
Oak Harbor, OH 43449-9760

Director, Ohio Department of Commerce
Division of Industrial Compliance
Bureau of Operations & Maintenance
6606 Tussing Road
P.O. Box 4009
Reynoldsburg, OH 43068-9009

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
Suite 210
2443 Warrenville Road
Lisle, IL 60532-4352

Resident Inspector
U.S. Nuclear Regulatory Commission
5503 North State Route 2
Oak Harbor, OH 43449-9760

Stephen Helmer
Supervisor, Technical Support Section
Bureau of Radiation Protection
Ohio Department of Health
35 East Chestnut Street, 7th Floor
Columbus, OH 43215

Carol O'Claire, Chief, Radiological Branch
Ohio Emergency Management Agency
2855 West Dublin Granville Road
Columbus, OH 43235-2206

Zack A. Clayton
DERR
Ohio Environmental Protection Agency
P.O. Box 1049
Columbus, OH 43266-0149

State of Ohio
Public Utilities Commission
180 East Broad Street
Columbus, OH 43266-0573

Attorney General
Office of Attorney General
30 East Broad Street
Columbus, OH 43216

President, Board of County
Commissioners of Ottawa County
Port Clinton, OH 43252

President, Board of County
Commissioners of Lucas County
One Government Center, Suite 800
Toledo, OH 43604-6506

The Honorable Dennis J. Kucinich
United States House of Representatives
Washington, D.C. 20515

The Honorable Dennis J. Kucinich
United States House of Representatives
14400 Detroit Avenue
Lakewood, OH 44107

Joseph J. Hagan
President and Chief Nuclear Officer
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

David W. Jenkins, Attorney
FirstEnergy Corporation
Mail Stop A-GO-15
76 South Main Street
Akron, OH 44308

Danny L. Pace
Senior Vice President, Fleet Engineering
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

Manager, Fleet Licensing
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-2
76 South Main Street
Akron, OH 44308

Director, Fleet Regulatory Affairs
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-2
76 South Main Street
Akron, OH 44308

cc:

Jeannie M. Rinckel
Vice President, Fleet Oversight
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

Richard Anderson
Vice President, Nuclear Support
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

James H. Lash
Senior Vice President of Operations
and Chief Operating Officer
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

REQUEST FOR ADDITIONAL INFORMATION

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

In reviewing the FirstEnergy Nuclear Operating Company's (FENOC's) submittal dated April 12, 2007, related to FENOC's license amendment request (LAR) for measurement uncertainty recapture power uprate, for the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS), the Nuclear Regulatory Commission (NRC) staff has determined that the following information is needed in order to complete its review:

1. Engineering Instrumentation & Controls Branch (EICB): Enclosure 1 stated in Section 5.2 that the limiting safety system setpoints (LSSS) for the reactor protection system high flux functional unit is the limiting trip set point specified in updated final safety analysis report (UFSAR) technical requirements manual and the allowable value relationship to the setpoint methodology and testing requirements in the technical specifications is documented in the setpoint calculation which is maintained as part of plant records. Please submit this calculation assuring that the calculation documents (including sample calculation) the methodology used for establishing the limiting set point or the nominal set point and the limiting acceptable values for the as-found and as-left set point as measured in periodic surveillance testing. This calculation should also indicate the related analytical limit and other limiting design values (and the sources of these values) for the high flux functional unit LSSS.

2. EICB: Section 1.1 in Enclosure 2 states that the Leading Edge Flow Meter (LEFM) flow meters (one in each of the two steam generator feedwater flow headers) were calibrated at the Alden Research Laboratory using the plant's current piping configuration and <u>variations</u> of the plant's configuration. Please explain those variations of the plant's configuration and submit the calibration report.

3. EICB: Section 2.1(2) in Enclosure 3 states that the correspondence between the plant computer IDs and the variables used in CTPA was not formally provided to AREVA NP and, therefore, the information is assumed. The heat balance uncertainty calculations in enclosure 3 was performed by AREVA NP using the assumed values. Please explain the validity of the assumed values in this calculation and how the actual values will affect the heat balance uncertainty.

4. EICB: Section 1.1.E in Enclosure 2 states that Camron Measurement Systems (formerly Caldon Inc.) has performed an evaluation of the uncertainty involved in replacing LEFM Check Plus transducers in the field. Please submit this calculation.

5. Component Performance & Training Branch: Section IV.1.A.ix, Safety-Related Valves - Describe whether the design bases of safety-related valves have been evaluated for the pressure changes due to the measurement uncertainty recapture power uprate. Also explain whether pressure locking effects on safety-related power-operated valves have been evaluated for any change in differential pressure per the recommendations in Generic Letter (GL) 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves."

6. Fire Protection-1: LAR, Enclosure 2, Attachment A, "D-B MUR [measurement uncertainty recapture] Summary Report," Section II, "Accidents and transients for which the existing analyses of record bound plant operation at the proposed uprated power level," mentions safe-shutdown fire analysis. However this section does not discuss the impact of measurement uncertainty recapture power uprate on the fire protection system(s). Clarify whether this request involves changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe-shutdown capability in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix R. Provide the technical justification for whether and, if so, why, existing analyses bound any impact on accidents or transients resulting from any changes.

7. Fire Protection-2: The NRC staff notes that LAR, Enclosure 2, Attachment A, "D-B MUR Summary Report" Section III, "Accidents and transients for which the existing analyses of record do not bound plant operation at the proposed uprated power level," does not include any discussion regarding changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe shutdown capability in accordance with Appendix R. Clarify whether this request involves changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe-shutdown capability in accordance with 10 CFR Part 50, Appendix R. Provide the technical justification for whether and, if so, why, existing analyses do not bound any impact on accidents or transients resulting from any changes.

8. Fire Protection-3: Section VI, "Post Fire Safe Shutdown Capability," of the NRC safety evaluation report dated May 30, 1991, on page 29 states that:

   The NRC staff's conclusion is also based on the statements made by the licensee in its letter dated June 6, 1988, that the capability to return the pressurizer level to within prescribed instrument indication range, and to restore other process variables to within the range predicted by a loss of offsite power, will be preserved. In addition, the licensee states that the core will not be uncovered and fission product boundary integrity will not be affected during the postulated transient conditions.

   The NRC criteria which is applicable to the DBNPS post-fire safe-shutdown is contained in Sections III.G and III.L of Appendix R to 10 CFR Part 50, in GL 81-12, "Fire Protection Rule (45 FR 76602, November 19, 1980)," and its subsequent clarification in GL 86-10, "Implementation of Fire Protection Requirements." The NRC staff requests the licensee to verify whether the above conclusion is valid at an increased reactor power level of 2817 megawatts thermal (MWt), 1.63% above the currently licensed power level of 2772 MWt.

9. Vessels & Internals Integrity Branch (CVIB): Section IV.1.C.i, "Pressurized Thermal Shock (PTS)," of Enclosure 2 to the submittal dated April 12, 2007, indicates that the MUR power uprate projected fluence at 32 effective full power years (EFPY) for the limiting reactor vessel material, Weld WF-182-1, is 1.124×10^{19} n/cm^2 (E>1.0 MeV). However, Section IV.1.C.v, "Effect on Low Upper Shelf Energy," of Enclosure 2 indicates that the MUR projected inner-diameter (ID) fluence at 32 EFPY for this material is 1.02×10^{19} n/cm^2 (E>1.0 MeV). Please explain the discrepancy between these two ID fluence values.

10. CVIB: Section IV.1.A.viii, "Pressurizer Structural Evaluation," of Enclosure 2 to the submittal indicates that your pressurizer structural evaluation determined that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. What temperature change (e.g., hot leg and cold leg temperature change) do you refer to? How does this temperature change affect the most critical transient that was used in the existing pressurizer integrity analysis? Identification of new pressurizer insurges in recent years has caused reevaluation of pressurizer integrity for several pressurized-water reactor (PWR) plants. Confirm that you have considered appropriate pressurizer in-surges in your design transients.

11. CVIB: Enclosure 2 to the submittal provides very little information regarding your reactor vessel (RV) internals structural evaluation. Table Matrix-1 of NRC RS-001, Revision 0, "Review Standard for Extended Power Uprates," provides the NRC staff's basis for evaluating the potential for extended power uprates to induce aging effects on RV internals. Depending on the magnitude of the projected RV internals fluence, Table Matrix-1 may be applicable to the MUR application. In the Notes to Table Matrix-1, the NRC staff states that guidance on the neutron irradiation-related threshold for irradiation-assisted stress corrosion cracking (SCC) for PWR RV internal components are given in BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," and WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals." The Notes to Table Matrix-1 state that for thermal and neutron embrittlement of cast austenitic stainless steel, SCC, and void swelling, licensees will need to provide plant-specific degradation management programs or participate in industry programs to investigate degradation effects and determine appropriate management programs. Discuss your management of the above-mentioned aging effects on RV internals in light of the guidance in BAW-2248A and WCAP-14577, Revision 1-A. Please also confirm whether you have established an inspection plan to manage the age-related degradation in the DBNPS RV internals, or whether you have participated in the industry's initiatives on age-related degradation of PWR RV internals.

12. SG Tube Integrity & Chem. Engineering Branch (CSGB): Confirm that the steam generators (SG) will continue to satisfy all original design criteria under power uprate conditions. In addition, confirm that your analysis addresses the current condition of your SGs (e.g., plugs, tube repairs, loose parts, etc.) and addresses flow induced vibration.

13. CSGB: Provide confirmation that your SG tube plugging limit is still appropriate for power uprate conditions, given the guidance in Regulatory Guide 1.121, Bases for Plugging Degraded PWR Steam Generator Tubes.

14. CSGB: Confirm that the coating qualification temperature and pressure profile used to qualify the original maintenance Service Level I coatings continues to bound the design basis accident temperature and pressure profile under power uprate conditions.

15. CSGB: Please confirm the following regarding the SG blowdown system:

    a. That you considered whether the additional operating time due to the power uprate will result in system components being more susceptible to flow accelerated corrosion (FAC).

b. That your current evaluation of the SG blowdown system under power uprate conditions considered the effect of a potential increase of impurities in the SG water.

c. That any change to the inlet pressure of the SG blowdown system is still inside the range of operating parameters for the power uprate.

16. CSGB: You indicated that “the predicted increases in maximum component wear rates and reductions in service lives can be managed by the DBNPS FAC program.” Discuss how significant the increases in wear rates and reductions in service lives are for the power uprate conditions. In addition, discuss any changes made to DBNPS FAC program (i.e., criteria used for selecting components for inspection following the power uprate, criteria for repair and replacement, increased inspection scope, etc.) due to power uprate conditions. Also, identify the systems that are expected to experience the greatest increase in wear as a result of the power uprate. Discuss whether inspections will be performed to assess wear prior to entering power uprate conditions.

17. CSGB: Provide confirmation that your evaluation for the chemical and volume control system demonstrates that the conditions for the power uprate are bounded by the existing conditions (thermal performance, letdown and makeup requirements, etc.).

18. Reactor Systems Branch (SRXB): Please provide a copy of Caldon, Inc. Engineering Report: “Bounding Uncertainty Analysis for Thermal Power Determination at Davis Besse Nuclear Power Station Using LEFM ✓+ System,” July 2004.

19. SRXB: In the letter dated March 8, 2007, from Ed Madera, Cameron Measurement Systems Sr. Project Engineer, to Tim Laurer, Nuclear Staff Engineer, DBNPS, it is stated that Cameron proposed to provide a revised analysis that reflects the uncertainty associated with transducer replacement within 90 days. Please provide a copy of that information when it becomes available, or in your response to this RAI.

20. SRXB: Please provide a description and drawings that illustrate the feedwater piping configuration from the outlet of the feedwater pumps to the containment pressure boundary. Identify any perturbations in the piping wall that could affect the flow profile.

21. SRXB: Please provide a description and drawings of the Alden Laboratory test configuration used for the plant’s current piping configuration and variations of the plant’s configuration. Identify any differences between the test and plant configurations. Reference I-10 from your submittal of April 12, 2007, may be provided to address part of this request to alleviate preparation of additional documentation.

22. SRXB: Please provide a summary of the Alden Laboratory test results and application / comparison of those test results to plant operation. Include a representative set of test data if not provided in Request 21, above.

23. SRXB: Please provide a summary of LEFM characteristics before and after replacement of the 32 transducers in June 2006. This summary should contain a comparison to other feedwater measurement instruments over a sufficient time span to enable a valid comparison of before and after characteristics.

24. SRXB: If an LEFM becomes inoperative, you plan to rely upon venturis for a short time that have been calibrated with the last valid LEFM data. If a venturi defouling event should occur during this time, an overpower condition could result. Please discuss this possibility.

25. SRXB: How are plant personnel qualified to perform maintenance and calibration of the LEFM system?

26. SRXB: Discuss the frequency of the listed preventive maintenance activities.

27. SRXB: The discussion on the calibration of the flow meters indicates that the meters were calibrated at the Alden Research Laboratory facility using the plant's current piping configuration and variations of the plant's configuration. Explain what in the configuration was varied and why.

28. SRXB: The submittal indicates that the failure of one transducer resulted in an alarm that would have caused the LEFM system to be removed from service. The submittal further indicates that, since this initial failure in June 2005, one additional transducer has failed.

    - Did the second failure result in an alarm also? If not, why not?

    - Are these the only two transducer failures that have occurred over the life of the system, or have other transducer failures occurred?

29. SRXB: Section VIII.6.3.2 of th AREVA attachment indicates non-core sources of heat addition to the RCS power, then states, "A value of 17 MWt has been used by Davis-Besse." The wording, "has been used by," is unclear to the NRC staff. Provide additional comment on the basis for selection of this value.

30. SRXB: Upon annunciation of a transducer failure, how long before the LEFM system is declared inoperable?

31. SRXB: Section 15.1.2 of the UFSAR does not provide a very detailed anticipated transient without scram (ATWS) analysis. In comparing section 15.1.2 with the submittal, Section II.2, it appears that the maximum pressure criterion for an ATWS event is 3200 psig, whereas in the UFSAR, a safety limit is established as 2750 psig. The UFSAR does not discuss a maximum pressure criterion. Please explain the derivation of the maximum pressure criterion, its relationship to the American Society of Mechanical Engineers Code Section III pressure limit (2750 psig), and what pressures are predicted in an ATWS scenario.

32. SRXB: Regarding the Control Rod Assembly Misalignment analysis, explain why a power level of 102 percent (2966MWt) was selected for analysis, and how that differs from the assumptions of the original analysis. Why was this power level selected instead of the 102 percent generally selected for the remaining transients? In more detail, explain why this analysis is bounded by the analysis of record. Address any significant differences in peak pressure, peak temperature, and maximum reduction in DNBR margin, and any changes in the sequence of events.

33. Electrical Engineering Branch (EEEB): Provide a detailed comparison of existing ratings with uprated ratings and the effect of the power uprate on the following equipment:

- main generator rating and power factor
- isophase bus
- main power transformer
- unit auxiliary/startup transformer
- main generator breaker

34. EEEB: Does the power uprate affect any ac distribution system loads? If so, provide a list of loads affected by the power uprate change.

35. EEEB: Attachment A of the LAR refers to "Davis-Besse Stability Study for FirstEnergy Corporation" (ADAMS No. ML020640288). The transient stability study assumed a 10 percent increase in gross power output, which is significantly higher than the proposed increase of 1.63 percent. The study concluded that for two of the fourteen contingencies analyzed the system response varied or was unstable. A three-phase fault at the Bayshore 345 kiloVolt (kV) bus, Contingency 4, resulted in unstable system responses for the uprated system but stable conditions for the existing ratings. A three-phase fault at DBNPS circuit breaker 34564, Contingency 8, resulted in unstable system response. The study states, "If the Davis-Besse uprate occurs, additional analysis is recommended to determine methods to improve system stability [for Contingencies 4 and 8]." Have additional analyses been performed to evaluate improving system stability for Contingencies 4 and 8? If so, what actions are being taken as a result of the additional analyses?

36. EEEB: Provide justification that the DBNPS Stability Study completed in May 2000 bounds the current grid conditions. Specifically, since the results of the stability analysis are based on 1999/2000 summer peak load conditions, describe the impact on grid stability when using current summer peak loads.

37. EEEB: The DBNPS Stability Study indicates that with a 10 percent increase in gross power output, the change in power factor reduces the unit's reactive power capability by 67 mega volt ampere reactive (MVAR). For the current uprate of 1.63 percent, please address and discuss the following:

- Identify the nature and quantity of MVAR support necessary to maintain post-trip loads and minimum voltage levels. Address how the power uprate affects MVAR support.

- Identify what MVAR contributions DBNPS is credited by the transmission system operator (TSO) to support the grid. Address how the power uprate changes the MVAR contributions credited by the TSO.

- Address the compensatory measures taken to compensate for the depletion of the nuclear unit MVAR capability on a grid-wise basis due to this power uprate.

- Provide an evaluation of the impact of any MVAR shortfall listed in part C on the ability of the offsite power system to maintain post-trip voltage levels and to supply power to safety buses during peak electrical demand periods. The subject evaluation should document any information exchanges between the TSO and DBNPS on this matter.