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

March 17, 2011

Mr. Barry S. Allen
Vice President, Davis-Besse Nuclear Power Station
FirstEnergy Nuclear Operating Company
5501 North State Route 2
Oak Harbor, OH 43449

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON THE REACTOR VESSEL SURVEILLANCE AGING MANAGEMENT PROGRAM AND TIME-LIMITED AGING ANALYSES FOR NEUTRON EMBRITTLEMENT FOR THE REVIEW OF THE DAVIS-BESSE NUCLEAR POWER STATION (TAC NO. ME4640)

Dear Mr. Allen:

By letter dated August 27, 2010, FirstEnergy Nuclear Operating Company (FENOC), submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 (10 CFR Part 54) for renewal of Operating License NPF-3 for the Davis-Besse Nuclear Power Station (DBNPS). The staff of the U.S. Nuclear Regulatory Commission (NRC or the staff) is reviewing this application in accordance with the guidance in NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants." During its review, the staff has identified areas where additional information is needed to complete the review. The staff's requests for additional information are included in the Enclosure. Further requests for additional information may be issued in the future.

Items in the enclosure were discussed with Cliff Custer, of your staff, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me by telephone at 301-415-2277 or by e-mail at brian.harris2@nrc.gov.

Sincerely,

Brian K. Harris, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosure:
As stated

cc w/encl: Listserv

REQUEST FOR ADDITIONAL INFORMATION
DAVIS-BESSE NUCLEAR POWER STATION
LICENSE RENEWAL APPLICATION

**RAI B.2.35-1**

License Renewal Application (LRA) Section B.2.35 states that "[c]apsule TE1-C contains the Davis-Besse limiting material and has been exposed to fluence slightly above the 60-year projected fluence for the Davis-Besse plant."

Please state whether the limiting material referred to in this statement is Upper Shell to Lower Shell Circumferential Weld WF-182-1. Please state the fluence value for Capsule TE1-C.

**RAI B.2.35-2**

LRA Section B.2.35 states that Capsule TE1-E was removed from the reactor vessel and has been discarded.

Please explain why this surveillance capsule was discarded.

LRA Section 4.2.2 - USE Evaluation

**RAI 4.2.2-1**

LRA Section 4.2.2.1 states that, as no initial USE data is available for the reactor vessel (RV) beltline welds, plant operation through 32 EFPY (40 years) was justified based on an equivalent margins analysis (EMA). LRA Section 4.2.2.1 provides references for the subject EMA in LRA Section 4.8, References 4.8-2 and 4.8-3. LRA Section 4.2.2.3 states that the equivalent margins analysis (EMA) for the limiting beltline weld (WF-182-1) is projected to satisfy the acceptance criteria of the ASME Code, Section XI (the Code), Appendix K through the period of extended operation (PEO) (52 EFPY).

a. Are the existing criteria for minimum acceptable USE developed in References 4.8-2 and 4.8-3 (using ASME Code, Section XI, Appendix K EMA procedures) valid for demonstrating RV beltline weld acceptability through 52 EFPY, based on the calculations of the projected percentage decrease in USE for 52 EFPY, as listed in LRA Table 4.2-2?

b. If the existing criteria for minimum acceptable USE developed in References 4.8-2 and 4.8-3 are not valid for demonstrating RV beltline weld acceptability through 52 EFPY, then please provide the reports documenting the EMA calculations for demonstrating that all RV beltline welds, including the limiting beltline weld (WF-182-1), will satisfy the requirements of 10 CFR Part 50, Appendix G for equivalent margins against ductile fracture through the PEO (52 EFPY).

**RAI 4.2.2-2**

LRA Table 4.2-2 lists the initial USE for all RV beltline welds (Linde 80) as 70 ft-lbs. Please explain how the 70 ft-lb initial USE value for these welds was obtained or derived (i.e., from a conservative estimate based on a statistically significant sample of existing Charpy USE data for this type of weld, or other method), given the statement in LRA Section 4.2.2.1 that no initial USE data is available for the RV beltline welds. If the 70 ft-lb initial USE value is a conservative estimate based on a statistically significant sample of existing Charpy USE data for this type of weld, please state whether the EMA calculation reports (References 4.8-2 and 4.8-3) included the statistical analysis of the Charpy USE data for Linde 80 welds. Otherwise, please provide the statistical analysis of the Charpy USE data for this type of weld.

**RAI 4.2.2-3**

LRA Section 4.2.2.2 states that Regulatory Position (RP) 2.2 of Regulatory Guide 1.99, Rev. 2 was used to calculate 52 EFPY USE values for weld WF-182-1 and forging BCC 241 using surveillance data. Please state whether the 52 EFPY USE values (2$^{nd}$ line entry for 52 EFPY USE in Table 4.2-2 for each component) for these materials are based on two credible sets of USE surveillance data for these materials.

LRA Section 4.2.6 – Intergranular Separation/Underclad Cracking

**RAI 4.2.6-1**

Please provide a reference for the report documenting the detailed analyses for demonstrating that the postulated underclad cracks in the Davis-Besse RV SA-508, Class 2 forging materials are acceptable for the period of extended operation.

LRA Section 4.2.7 – Reduction in Fracture Toughness of RV Internals

**RAI 4.2.7-1**

The staff notes that cast austenitic stainless steel (CASS) components are susceptible to reduction in fracture toughness due to the synergistic effects of both neutron embrittlement and thermal embrittlement. The LRA aging management review (AMR) results for the RV internals (LRA Table 3.1.2-2) lists a number of CASS RV internal components.

a. Will these CASS RV internals components be screened for susceptibility to thermal embrittlement based on ferrite content, molybdenum content, and casting method under the Davis-Besse PWR Reactor Vessel Internals Aging Management Program (AMP)?

b. Will CASS RV internals components determined to be susceptible to thermal embrittlement (based on ferrite content, molybdenum content, and casting method) receive supplemental examinations and/or component specific evaluations of reduction in fracture toughness (due to the synergistic effects of both neutron embrittlement and thermal embrittlement) under the Davis-Besse PWR Reactor Vessel Internals AMP?

LRA Section 4.7 – Other Plant-Specific TLAAs

LRA Section 4.7.4 – High Pressure Injection/Makeup Nozzle Thermal Sleeves

LRA Section 4.7.4 states that the high pressure injection (HPI) makeup flow path was re-routed from HPI/makeup nozzle A-1 to nozzle A-2 during the Cycle 6 refueling outage (1990) as one of the corrective actions for the subject failed thermal sleeve. LRA Section 4.7.4 then states that fracture mechanics analysis of thermal sleeve life under various makeup flow cycling conditions predicted a thermal sleeve lifetime exceeding 20 eighteen-month operating cycles under current makeup flow control conditions. LRA Section 4.7.4 stated that, since that analysis, Davis-Besse had an extended (approximately two year) Cycle 13 refueling outage, converted to a 24-month fuel cycle, and performed a measurement uncertainty recapture power uprate. The corresponding predicted end-of-life for the HPI/makeup nozzle thermal sleeve is approximately 2022, based on the predicted number of makeup thermal cycles.

**RAI 4.7.4-1**

The staff notes that Davis-Besse has committed to replacing all four makeup nozzle thermal sleeves prior to the beginning of the period of extended operation. Based on the above discussion, please state which specific HPI/makeup nozzle thermal sleeves (thermal sleeves for HPI/makeup nozzle A-2 or other nozzle thermal sleeves) were analyzed as discussed above.

**RAI 4.7.4-2**

Please provide a reference for the subject thermal sleeve fracture mechanics analysis.

LRA Section 4.2.5 – Inservice Inspection – Fracture Mechanics Analyses

4.7.5.1 – Reactor Coolant System Loop 1 Cold Leg Drain Line Weld Overlay Repair

**RAI 4.7.5.1-1**

LRA Section 4.2.5.1 states that the applicant performed a full structural overlay repair for an axial indication found on the Reactor Coolant System Loop 1 cold leg drain line during the

Cycle 14 refueling outage. The structural weld overlay of the cold leg drain nozzle was designed consistent with the requirements of the ASME Code, Section XI; ASME Code Case N-504-2; ASME Code, Section XI, non-mandatory Appendix Q; and was supplemented by additional design considerations specific to the unique nature of the geometry and materials of the cold leg drain nozzle-to-elbow weld. The overlay is designed as a full structural overlay that assumes the as-found flaw propagates to a 100% through-wall 360-degree crack, rather than performing a crack growth analysis of the as-found flaw. The fatigue analysis for the repaired configuration conservatively estimated cycles for 60 years at 1.5 times the original design cycles.

Please provide a reference for the fatigue analysis of the repaired configuration discussed above, if this fatigue analysis is not referenced elsewhere in the LRA.

4.7.5.2 – Once-Through Steam Generator 1-2 Flaw Evaluations

**RAI 4.7.5.2-1**

LRA Section 4.7.5.2 discusses a number of flaws that were discovered in the steam generator 1-2 shell base material and shell welds during the Cycle 5 refueling outage (May 1988). The staff requests the following information concerning these flaws and the analyses performed for these flaws.

a. How many flaw indications were found in total (in 1988) that did not pass the initial ASME Code, Section XI, IWB-3500 screening criteria?

b. Were these flaws determined to be the result of service-induced degradation or fabrication defects?

c. Have the components with the flaws received subsequent/supplemental examinations, in accordance with ASME Code, Section XI requirements since May 1988?

d. When is the next inservice examination scheduled for the components with the flaws?

e. Have the flaw dimensions increased since discovery in 1988? If so, were the flaws re-analyzed in accordance with ASME Code, Section XI, IWB-3600 requirements based on the new flaw dimensions?

f. Are the existing flaw growth analyses for the subject flaws bounded by the projected number of thermal cycles for the period of extended operation? If not, the staff requests that the applicant provide a specific license renewal commitment to re-evaluate the subject flaws (per IWB-3600) for the period of extended operation, based on the results of the next scheduled inservice examination of the components with the flaws.

g. No flaw evaluation reports are referenced. Please provide references to reports documenting IWB-3600 analytical evaluations of the subject flaws. Were these reports previously submitted to the NRC?

March 17, 2011

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Vice President, Davis-Besse Nuclear Power Station
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Sincerely,
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Brian K. Harris, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosure:
As stated

cc w/encl: Listserv

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Letter to Barry S. Allen from Brian K. Harris dated March 17, 2011

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