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

August 11, 2011

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 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 submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 for renewal of Operating License NPF-3 for the Davis-Besse Nuclear Power Station. 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 request 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 301-415-2946 or by e-mail at Samuel.CuadradoDeJesus@nrc.gov.

Sincerely,

Samuel Cuadrado-De Jesús, 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

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

RAI B.2.34-2

Background:

In its response to request for additional information (RAI) B.2.34-1, FirstEnergy Nuclear Operating Company (the applicant) stated that according to the certificate of material test report (CMTR) for the reactor head closure studs, the actual measured yield strength varied from 151 to 159 ksi, and the tensile strength varied from 166 to 171 ksi. The applicant also stated that its reactor head stud material is SA-540, Grade B-23 and that as provided in Regulatory Guide (RG) 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," this material when tempered to a maximum tensile strength of 170 ksi, is relatively immune to stress corrosion cracking (SCC). The applicant proposes to enhance the Reactor Head Closure Studs Program to preclude the future use of replacement closure stud bolting fabricated from material with actual measured yield strength greater than or equal to 150 ksi, except for use of the existing spare reactor head closure stud bolting.

The "preventive actions" program element of generic aging lessons learned (GALL) aging management program (AMP) XI.M3, "Reactor Head Closure Studs Bolting," references the guidance in RG 1.65 and NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants."

Issue:

License renewal application (LRA) Section B.2.34 states that the Reactor Head Closure Program is an existing program that, with enhancements, will be consistent with GALL AMP XI.M3. All of the applicant's reactor head closure studs were fabricated from material with measured yield strength above 150 ksi and some of the furnished materials have a measured tensile strength above 170 ksi. The U.S. Nuclear Regulatory Commission (the staff) noted that this is an exception to the "preventive actions" program element of GALL AMP XI.M3, which recommends using bolting material for closure studs with actual measured yield strength less than 150 ksi to reduce susceptibility to SCC.

Request:

1) Revise the appropriate sections of the LRA to reflect the use of reactor head closure studs with measured yield strength above 150 ksi as an exception to GALL AMP XI.M3.

2) Address the exception to the "preventive actions" element for using closure stud material with greater susceptibility to SCC. Justify the adequacy of the Reactor Head Closure Studs Program to manage cracking due to SCC of high-strength bolting material. As part of the justification, describe how the program manages the potential exposure of closure bolting to borated water and other potential contaminants that may initiate SCC of the reactor head closure bolting studs and components.

RAI 3.1.2.2-3

Background:

In Request 3 of RAI 3.1.2.2-2 issued by letter dated June 20, 2011, the staff requested that the applicant describe the functional groups for the following two components that are addressed in LRA Table 3.1.2-2: (1) core support assembly (CSA) vent valve body, and (2) plenum cylinder reinforcing plate. The staff also requested that if existent, the applicant describe their link relationships (such as primary/expansion link) with other components. In addition, the applicant was requested to describe the inspection method, including the inspection frequency, for the components and the technical basis for the applicant's aging management methods.

In its response dated July 22, 2011, the applicant stated that in Topical Report MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," the reactor internals were assigned to one of the following four functional groups: Primary, Expansion, Existing Programs, and No Additional Measures components. The applicant also stated that the link relationships are consistent with that provided in Tables 4-1 and 4-4 of MRP-227, Rev. 0. The applicant further stated that the inspection frequency and method for the primary and expansion components are provided in Tables 4-1 and 4-4 of MRP-227, Rev. 0. In comparison, the revised LRA Table 3.1.2-2 in response to RAI 3.1.2.2-2 does not include an AMR item to manage loss of fracture toughness of the cast austenitic stainless steel (CASS) CSA vent valve body and plenum cylinder reinforcing plate.

In its review, the staff noted that GALL Report, Rev. 2, item IV.B4.RP-382 recommends GALL AMP XI.M1, "ASME [American Society of Mechanical Engineers Boiler and Pressure Vessel Code] Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," to manage cracking or loss of material due to wear of core support structure components; however, the LRA does not address this item. The staff also noted that Section 5.4.4 of the applicant's Technical Specifications requires that it should be verified by visual inspection every 24 months that the vent valve body exhibits no abnormal degradation. In addition, the staff noted that Section 3.2.3, Table 3-2 and Section 4 of Topical Report BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," indicate that reduction of fracture toughness due to thermal aging embrittlement is applicable to reactor vessel internal vent valve bodies.

In its review, the staff also noted that the revised LRA Table 3.1.2-2 submitted by letter dated July 22, 2011, does not address the following GALL Report Rev. 2 items:
(1) items IV.B4.RP-236 and IV.B4.RP-237 for the components with no additional measures and

(2) items IV.B4.RP-238 and IV.B4.RP-239 for the inaccessible locations of the reactor vessel internals.

Issue:

In its response to RAI 3.1.2.2-2, the applicant indicated that the applicant's aging management methods for the plenum cylinder reinforcing plate and vent valve body are described in MRP-227 Tables 4-1 and 4-4. However, the staff noted that MRP-227 Tables 4-1 and 4-4 referenced in the applicant's response do not clearly address information regarding: (1) the functional groups, (2) the link relationships, or (3) the inspection method, including the frequency, specified for the CSA vent valve body and plenum cylinder reinforcing plate. In addition, the revised LRA Table 3.1.2-2 in response to RAI 3.1.2.2-2 does not address an AMR line item to manage loss of fracture toughness of these CASS components.

In its review, the staff also found a need to clarify the following items: (1) why LRA Table 3.1.2-2 does not address GALL Report, Rev. 2, items IV.B4.RP-382, IV.B4.RP-236, IV.B4.RP-237, IV.B4.RP-238 and IV.B4.RP-239, (2) whether or not GALL Report, Rev. 2, item IV.B4.RP-382 is applicable to the plenum cylinder reinforcing plate and vent valve body, and (3) why LRA Table 3.1.2-2 does not address an AMR item for aging management of loss of fracture toughness of the vent valve body even though applicant's Technical Specifications require visual inspections of the component to ensure no abnormal degradation and Topical Report BAW-2248A indicates that reduction of fracture toughness due to thermal aging embrittlement is applicable to reactor vessel internal vent valve bodies.

Request:

1. Provide the justification as to why LRA Table 3.1.2-2 does not address the following GALL Report items for the components with no additional measures and inaccessible areas: GALL Report items IV.B4.RP-236, IV.B4.RP-237, IV.B4.RP-238 and IV.B4.RP-239

   In addition, describe the applicant's operating experience to clarify whether or not the accessible areas of the applicant's components have indicated aging effects that need management.

2. Provide the justification as to why LRA Table 3.1.2-2 does not address GALL Report, Rev. 2, item IV.B4.RP-382 that recommends GALL AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," to manage cracking or loss of material of core support structure. In addition, clarify whether or not this item for the core support structure is applicable to the plenum cylinder reinforcing plate and vent valve body.

3. Provide the justification as to why LRA Table 3.1.2-2 does not address an AMR item to manage loss of fracture toughness of the CASS vent valve body even though applicant's Technical Specifications require visual inspections of the component to ensure no abnormal degradation and Topical Report BAW-2248A indicates that reduction of facture toughness is applicable to the internal valve bodies.

4. Provide the information regarding: (1) the functional groups, (2) the link relationships (if existent), and (3) the inspection method including the frequency used to manage loss of fracture toughness of the CSA vent valve body and plenum cylinder reinforcing plate. As part of the response, provide the technical basis to demonstrate that these aging management methods are adequate to manage loss of fracture toughness of the components.

   If the functional group of the components is Existing Programs or No Additional Measures group, provide the method and frequency of the existing inspections specified for the CASS components.

RAI 4.3.2.3.2-1 - (Supplement)

Background:

By letter dated June 22, 2011, the applicant responded to RAI 4.1-1 regarding cumulative usage factor (CUF) or I_t fatigue analyses for Class 1 valves. In its response to RAI 4.1-1, Request 1, Part A, the applicant identified 12 large bore Class 1 valves (i.e., valves with nominal pipe sizes in excess of 4-inches) that should have received CUF or I_t fatigue analyses in accordance with the design codes (i.e., 1971 or more recent Editions of the ASME Code Section III, or the 1968 Edition of the Draft ASME Pump and Valve Code for Nuclear Power Plants). The applicant provided Commitment No. 46 to complete the following, prior to April 22, 2015:

> FENOC commits to perform a fatigue evaluation in accordance with the requirements of the ASME Code of record for the Davis-Besse Class 1 valves that are greater than 4 inches nominal pipe size. The applicable valve identification numbers are CF28, CF29, CF30, CF31, DH76, DH77, DH11, DH12, DH1A, DH1B, DH21, and DH23.

LRA Section 4.3.2.3.2, as amended by letter dated June 22, 2011, states that the fatigue analyses for these 12 referenced large bore Class 1 valves are as TLAAs and are dispositioned in accordance with Title 10 of the *Code of Federal Regulations* 54.21(c)(1)(iii), that the effects of fatigue on Class 1 valves greater than 4 inches diameter nominal pipe size will be managed for the period of extended operation by the Fatigue Monitoring Program. LRA Section 4.3.2.3.2 also states that the issue with the missing CUF or I_t calculations for the 12 referenced large bore Class 1 valves has been entered into the applicant's Corrective Actions Program.

Issue:

The information provided by the applicant in letter of June 22, 2011, did not provide information regarding whether the applicant had any ASME Code, Section III NB-3222.4(d) fatigue waiver assessments (or equivalent waiver assessments permitted by the 1968 Draft ASME Pump and Valve Code) for the 12 large bore Class 1 valves referenced in Commitment No. 46. Therefore, the staff requests additional information regarding whether fatigue calculations are required for these valves.

The staff is concerned that without the CUF or I_t analyses or an appropriate fatigue waiver or exemption for these 12 large bore Class 1 valves, the staff would not be able to evaluate whether the aging effects will be appropriately managed by the commitment.

<u>Request</u>:

Provide justification for not having the analyses for staff review as part of the LRA, or provide your appropriate fatigue waiver or fatigue exemption bases for not having such analyses.

August 11, 2011

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 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 submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54 for renewal of Operating License NPF-3 for the Davis-Besse Nuclear Power Station. 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 request for additional information are included in the enclosure. Further requests for additional information may be issued in the future.

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Sincerely,

/RA/

Samuel Cuadrado-De Jesús, 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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ADAMS Accession No.: ML11216A236 *concurrence via e-mail

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NAME	SFigueroa	SCuadrado	DMorey	SFigueroa	SCuadrado
DATE	08/8/2011	08/10/2011	08/10/2011	08/11/2011	08/11/2011

OFFICIAL RECORD COPY

Letter to Barry S. Allen from Samuel Cuadrado-De Jesús dated August 11, 2011

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE DAVIS-BESSE NUCLEAR POWER STATION (TAC NO. ME4640)

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