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

October 12, 2017

Mr. Mark B. Bezilla
Site Vice President
FirstEnergy Nuclear Operating Company
Mail Stop A-DB-3080
5501 N. State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 – TIME-LIMITED AGING ANALYSIS FOR THE REACTOR VESSEL INTERNALS LOSS OF DUCTILITY AT 60 YEARS (CAC NO. MF9126, EPID L-2017-LRO-0002)

Dear Mr. Bezilla:

By letter dated January 23, 2017 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML17026A004), as supplemented by letters dated March 23 and June 8, 2017 (ADAMS Package Accession Nos. ML17086A019 and ML17163A418, respectively), FirstEnergy Nuclear Operating Company (the licensee) submitted an evaluation for review and approval by the U.S. Nuclear Regulatory Commission (NRC) in response to Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse), license renewal commitment No. 54. Specifically, the licensee submitted its time-limited aging analysis (TLAA) for the reactor vessel internals loss of ductility at 60 years.

The NRC staff has completed its review of the TLAA and supplemental information provided by the licensee, and concludes that the licensee's TLAA for the loss of ductility of the reactor vessel internals at Davis-Besse is acceptable. Therefore, Davis-Besse license renewal commitment No. 54 has been fulfilled. The staff's review is documented in the enclosed safety evaluation.

If you have any questions regarding this matter, I may be reached at 301-415-1380.

Sincerely,

Blake Purnell, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosure:
Safety Evaluation

cc w/encl: Distribution via ListServ

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TIME-LIMITED AGING ANALYSIS FOR REACTOR VESSEL INTERNALS

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-346

1.0 INTRODUCTION

By letter dated January 23, 2017 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML17026A004), as supplemented by letters dated March 23 and June 8, 2017 (ADAMS Package Accession Nos. ML17086A019 and ML17163A418, respectively), FirstEnergy Nuclear Operating Company (FENOC, the licensee) submitted an evaluation for review and approval by the U.S. Nuclear Regulatory Commission (NRC) in response to Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse), license renewal commitment No. 54. Specifically, the licensee submitted its time-limited aging analysis (TLAA) for the reactor vessel internals (RVI) loss of ductility at 60 years. By email dated May 9, 2017 (ADAMS Accession No. ML17129A411), the NRC issued a request for additional information (RAI), which the licensee responded to in its June 8, 2017, letter (RAI response).

2.0 REGULATORY EVALUATION

2.1 Background

On December 8, 2015, the NRC issued Renewed Facility Operation License No. NPF-3 for Davis-Besse. The licensee made several commitments in its license renewal application that were incorporated into the Davis-Besse updated final safety analysis report (UFSAR) by License Condition 2.C(11), "License Renewal License Conditions." These commitments are documented in Appendix A of NUREG-2193, Supplement 1, "Safety Evaluation Report Related to the License Renewal of Davis-Besse Nuclear Power Station," dated August 2015 (ADAMS Accession No. ML16104A350).

The licensee's RVI program is based on the NRC-approved Electric Power Research Institute technical report MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," dated December 2011 (ADAMS Package Accession No. ML120170453). With respect to the licensee's RVI program, the NRC staff concluded in NUREG-2193, Supplement 1, that FENOC demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation, as required by Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 54.21(a)(3). In making this conclusion, the staff considered license renewal commitment No. 54, which states:

> In response to MRP-227-A Applicant/Licensee Action Item 8, update and submit for NRC review and approval an evaluation for the period of extended operation regarding the effect of irradiation on the mechanical properties and deformation limits of the RV [reactor vessel] internals that was evaluated for the current term of operation in Appendix E of Topical Report BAW-10008, Part 1, Revision 1, [1] supplemented by Davis-Besse U[F]SAR Appendix 4A.

The licensee's January 23, 2017, letter, as supplemented, provides the evaluation identified in this commitment.

2.2 Regulatory Requirements and Guidance

Paragraph 54.21(a)(3) of 10 CFR requires license renewal applications to include information which demonstrates that, for each structure and component subject to an aging management review per 10 CFR 54.21(a)(1), the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

Paragraph 54.21(c)(1) of 10 CFR requires a list of TLAAs, as defined in 10 CFR 54.3, "Definitions," to be provided with the license renewal applications. Paragraph 54.21(c)(1) further states that the applicant shall demonstrate that:

- (i) The analyses remain valid for the period of extended operation;
- (ii) The analyses have been projected to the end of the period of extended operation; or
- (iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

NRC Final License Renewal Interim Staff Guidance LR-ISG-2011-05, "Ongoing Review of Operating Experience" (ADAMS Accession No. ML12044A215) provides current guidance regarding the consideration of operating experience in license renewal applications. The guidance states, in part (page A-4), that: "an applicant should ensure that it has adequate processes to monitor and evaluate plant-specific and industry operating experience related to aging management to ensure that the [aging management programs] are effective in managing the aging effects for which they are credited."

NRC Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001 (ADAMS Accession No. ML010890301), describes methods and assumptions acceptable to the NRC staff for determining the pressure vessel neutron fluence. However, the guidance may not be appropriate for estimating neutron fluence for some RVI components.

---

[1] AREVA Document BAW-10008, Part 1, Revision 1, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake," June 1970. This will be referred to as BAW-10008 in this safety evaluation.

3.0 TECHNICAL EVALUATION

3.1 Licensee Evaluation

With the January 23 and March 23, 2017, letters, the licensee provided proprietary and nonproprietary versions of AREVA Inc. licensing report ANP-3542P, "Time-Limited Aging Analysis (TLAA) Regarding Reactor Vessel Internals Loss of Ductility for Davis-Besse Nuclear Power Station, Unit No. 1 at 60 Years" (the AREVA report). Section 4, "Inputs," of the AREVA report provided the inputs used in the loss-of-ductility evaluation described in Section 5, "Analysis," of the AREVA report. Table 4-2, "Stress Summary for Davis-Besse Reactor Internals (Davis-Besse UFSAR, Table 4.2-5)," of the report provided the results of a detailed stress analysis of the Davis-Besse RVI components under accident conditions. Table 4-2 identifies 22 stress analysis cases covering the different RVI components. Section 4 also provided the yield stress values at 600 degrees Fahrenheit (°F) (Table 4-1) for different types of materials used in RVI components based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III.

Section 5 of the AREVA report provided an evaluation methodology that used four categories to sequentially screen out all 22 stress analysis cases for RVI components by demonstrating that they do not have a loss-of-ductility concern. Category 1 screened out 13 cases where stress intensity values are less than the unirradiated ASME Code yield stress at 600 °F. Category 2 screened out one stress analysis case where the RVI component is already highly irradiated, such that the faulted stress intensity is below the irradiated yield stress and plasticity will not occur at 60 years or 52 effective full power years (EFPY). Category 3 screened out six stress analysis cases where the neutron fluence is low and embrittlement is negligible, such that loss of ductility is minimal or will not occur at 60 years or 52 EFPY and unirradiated ductility properties are still applicable. Category 4 contains the two analysis cases which were not screened out by Categories 1, 2, and 3. The assessment of these two cases is contained in Section 5.5 of the AREVA report, which determined there is not a loss-of-ductility concern for these cases. The AREVA report concludes, based on this analysis, that the loss of ductility for the Davis-Besse RVI components is acceptable and remains valid for a 60-year or 52-EFPY lifetime of the components.

3.2 Staff Evaluation

The NRC staff reviewed the inputs and the analysis for each of the four categories in the AREVA report. The staff considered its approval of the TLAA for RVI component loss of ductility at Arkansas Nuclear One, Unit No. 1 (ANO-1), which is documented in a May 28, 2015, safety evaluation (ADAMS Accession No. ML15139A113), in its review of the TLAA for Davis-Besse. ANO-1 is a similar reactor design to Davis-Besse and certain aspects of its loss-of-ductility methodology are relevant to Davis-Besse.

3.2.1 AREVA Report Inputs

Section 4 of the AREVA report states that the inputs to loss-of-ductility evaluation for the Davis-Besse RVI components are: (1) the unirradiated yield stress values at 600 °F based on the ASME Code, Section III, and (2) the results of the stress analysis for RVI components under accident conditions from the Davis-Besse UFSAR, Table 4.2-5.

Loss of ductility occurs in stainless steel components when they are exposed to high energy neutron fluence. The effect of irradiation on the mechanical properties and deformation limits for the RVI components was evaluated for 40 years in Appendix E of BAW-10008. The licensee concluded in its license renewal application that the TLAA reported in BAW-10008, Appendix E, is applicable to Davis-Besse. Section 2 of the AREVA report states that BAW-10008, Appendix E, identified that the core barrel, in the region near the flanges, is subjected to the maximum stress intensity where a loss of ductility would be detrimental.

The NRC safety evaluation for the ANO-1 review states that the stress intensity values presented in BAW-10008, Appendix E, were re-examined, and the location of highest stress intensity was found to occur at a different RVI component than listed in BAW-10008, Appendix E. The NRC staff determined that identification of the critical Davis-Besse RVI components in Table 4-2 of the AREVA report is consistent with the approval of the ANO-1 TLAA.

Table 4-1 of the AREVA report states that the yield stress value for alloy A-286 is based on three times the design stress intensity at 600 °F. In response to RAI-1, the licensee clarified that Appendix II of the 1968 Edition of the ASME Code, Section III, states that the stress values for bolting materials (ASME Code, Table N-422) are based on one-third of the minimum specified yield stress at room temperature or one-third of the yield stress at temperatures up to 800 °F. Therefore, the NRC staff determined that using three times the stress intensity as the yield stress at 600 °F is consistent with the ASME Code requirement for bolting materials.

3.2.2 AREVA Report Analysis – Category 1

Category 1 screened out the stress intensities of the RVI components that are less than the unirradiated yield stresses (AREVA Report, Table 4-1). This is acceptable because: (1) screening based on unirradiated yield stress is conservative and (2) plasticity will not occur at 60 years or 52 EFPY when the stress intensity of a RVI component is less than its irradiated yield stress. The NRC staff confirmed that there are 13 analysis cases which have no issue with loss of ductility at 60 years using the material information for each RVI component (AREVA report, Table 4-1, and RAI-2 response) and the stress intensity values for the 22 analysis cases (AREVA report, Table 4-2).

3.2.3 AREVA Report Analysis – Category 2

Category 2 screened out one stress analysis case where the RVI component is already highly irradiated, such that the faulted stress intensity is below the irradiated yield stress and plasticity will not occur at 60 years or 52 EFPY. Specifically, the licensee demonstrated that the stress intensity of one highly irradiated RVI component is less than the irradiated saturated yield stress. Section 5.2 of the AREVA report states that the expected neutron fluence exposures for the RVI components were reviewed, and only one component was determined to be highly irradiated.

The licensee's RAI-3a response summarizes the NRC-approved neutron fluence calculational methodology in BAW-2241P-A, Revision 1, "Fluence and Uncertainty Methodologies," December 1999 (public version at ADAMS Accession No. ML020930346), which was used to support the Davis-Besse neutron fluence evaluations for the reactor pressure vessel (RPV) beltline for 60 years and beyond. Similar evaluations[2] have shown that use of BAW-2241P-A, Revision 1, is also appropriate for estimating the neutron fluence for certain RVI components

[2] See the NRC safety evaluations for the staff's review of the ANO-1 and Oconee Nuclear Station (ADAMS Accession Nos. ML13045A489) TLAAs for RVI components loss of ductility.

above the core that are relevant to the Davis-Besse TLAA. The licensee also provided information regarding the estimated neutron fluence values for regions beyond the RPV beltline in the RAI-3a response.

In the RAI-3b response, the licensee identified the RVI component in Category 2 that required an irradiated yield stress margin assessment. Based on the location of this RVI component, the NRC staff determined that it is effectively within the 60-year RPV beltline where the neutron fluence calculational methodology in BAW-2241P-A, Revision 1, is fully applicable and expected to produce nominal fluence estimates with an uncertainty less than or equal to 20 percent. The licensee performed a bounding uncertainty analysis for this component which demonstrated that adequate irradiated yield stress margin will be maintained during the period of extended operation, even if nominal fluence estimates are reduced significantly. Therefore, the staff finds the neutron fluence calculational methodology, including the uncertainty estimate, acceptable for the Category 2 assessment.

Based on the above, the NRC staff determined that the estimated fluence value for the highly irradiated RVI component is credible. This value can be used in conjunction with Figure 5-1 of the AREVA report to arrive at the irradiated yield stress for comparison with the stress intensity listed Table 5-2. The licensee's conclusion that the loss of ductility for the highly irradiated RVI components is acceptable because the stress intensity for the component is less than the irradiated saturated yield stress with a significant margin.

3.2.4 AREVA Report Analysis – Category 3

Category 3 was used to screen out six stress analysis cases where the neutron fluence is low and embrittlement is negligible, such that loss of ductility is minimal or will not occur at 60 years or 52 EFPY and unirradiated ductility properties are still applicable. Specifically, Section 5.3 of the AREVA reports states that the cases were screened out if the decrease in uniform elongation for the RVI component at 572 °F and 752 °F met the 20 percent uniform elongation of irradiated material credited for 40 years (Appendix E of BAW-10008) and the 8.6 percent allowable strain (Appendix A of BAW-10008). The components which were screened out were all fabricated with Type 304 stainless steel.

The licensee's response to RAI-3c indicates that at the assumed critical neutron fluence for the period of extended operation there is sufficient margin to the 20 percent uniform elongation criterion for each of the Category 3 RVI components. For two of the RVI components in Category 3, there is significant margin between the nominal and assumed critical neutron fluence estimates. For these two components, the NRC staff expects this margin to compensate for any additional uncertainty not already addressed by use of a methodology consistent with RG 1.190.

The other RVI components in Category 3 have nominal neutron fluence values close to the critical neutron fluence value assumed in the uniform elongation analysis. The same fluence estimation method was used in the NRC-approved TLAA for RVI loss of ductility at ANO-1. In a January 26, 2015, letter (ADAMS Package Accession No. ML15028A495), the licensee for ANO-1 stated, in part (page 12 of Attachment 2):

> The flux synthesis methods employed for the ANO-1 fluence calculation are based on the methods described in BAW-2241P-A, [Revision 1]. The methods provide a reliable estimate for regions that are both (a) above the core's active fuel, as well as (b) regions that are far from the fuel–baffle – plate surface. The

> accuracy and precision of the methods was validated by a full-scale benchmark experiment performed at the Davis Besse reactor. The results of the benchmark verified that the methodology is unbiased and has a standard deviation less than 20%....

The letter for ANO-1 further indicates that the benchmark experiment results demonstrated acceptable uncertainty in the neutron fluence estimate for certain RVI components relevant to Davis-Besse.

The licensee did not use the maximum neutron fluence value for RVI components close to the critical neutron fluence value, which would have been conservative and bounding. However, the difference between the assumed and maximum neutron fluence values is expected to be covered by the remaining margin between the greater than 20 percent uniform elongation amount at the assumed critical neutron fluence value and the uniform elongation limit of 20 percent at a higher neutron fluence value shown in Figure 5-2 of the AREVA report. Therefore, the NRC staff finds it acceptable for the licensee to use the assumed, rather than the maximum, neutron fluence value for the Davis-Besse RVI components that are close to the critical fluence value.

Based on the above, the NRC staff concludes that BAW-2241P-A, Revision 1, provides acceptable neutron fluence estimates for all applicable Category 3 RVI components at Davis-Besse during the period of extended operation.

Based on the credibility of the neutron fluence values and the relationship between neutron fluence and uniform elongation (Figure 5-3 of the AREVA report), the NRC staff confirmed that six analysis cases met the ductility criteria in Category 3 for 60 years of operation. As discussed above, the same fluence estimation method was used in the NRC-approved TLAA for RVI loss of ductility at ANO-1, which was validated by full-scale benchmark experiments performed at Davis-Besse. Based on the staff's approval for ANO-1, the evaluation of the effect of strain rate on the uniform elongation of unirradiated Type 304 solution annealed stainless steel at 600 °F for Davis-Besse is also acceptable. Therefore, the staff determined that loss of ductility is minimal or will not occur at 60 years or 52 EFPY for the Davis-Besse RVI components related to the six analysis cases screened out in Category 3.

3.2.5 AREVA Report Analysis – Category 4

Only two analysis cases required further evaluation (Category 4) as they were not screened out by Categories 1, 2, or 3. For one RVI component, the licensee recalculated the faulted condition stresses considering asymmetric effects as a result of pipe-break loading. Considering asymmetric effects as a result of a pipe break for an RVI component sensitive to these effects is a more realistic approach, and, therefore, is acceptable.

For the other RVI component, which is not sensitive to the local asymmetric effects, the licensee credited leak-before-break by eliminating primary loop pipe breaks to recalculate the faulted condition loads. The principle of leak-before-break is that the piping will leak and actions will be taken before a break actual happens. Therefore, crediting leak-before-break of the primary loop piping to recalculate the faulted loads is appropriate. Eliminating pipe breaks from consideration for the large size primary loop reduced the faulted loading significantly for the RVI component.

Using the recalculated faulted loads, the licensee determined the stress intensities for the two RVI components in Category 4 were less than the unirradiated yield stress at 600 °F.

Therefore, the NRC staff determined that these two RVI components do not have loss-of-ductility concern at 60 years or 52 EFPY.

4.0 CONCLUSION

The NRC staff has reviewed the licensee's TLAA for the Davis-Besse RVI loss of ductility at 60 years or 52 EFPY, including the related supplements. The staff finds that:

- The licensee has projected the neutron fluence for the RVI components using an acceptable methodology that is consistent with RG 1.190.
- The licensee's evaluation of the deformation limits in BAW-10008, considering the change in tensile properties due to irradiation of the Type 304 solution annealed stainless steel material, is appropriate.
- The licensee's evaluation methodology of sequentially screening out RVI components that do not have a loss-of-ductility concern is acceptable.

Based on the above, the NRC staff concludes that the licensee's TLAA for loss of ductility of the RVI components at Davis-Besse is acceptable, and license renewal commitment No. 54 is fulfilled.

Principal Contributors: Simon Sheng, NRR
Amrit Patel, NRR

Date of issuance: October 12, 2017

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 – TIME-LIMITED AGING ANALYSIS FOR THE REACTOR VESSEL INTERNALS LOSS OF DUCTILITY AT 60 YEARS (CAC NO. MF9126, EPID L-2017-LRO-0002) DATED OCTOBER 12, 2017

DISTRIBUTION:
PUBLIC
LPL3 r/f
RidsACRS_MailCTR Resource
RidsNrrDorlLpl3 Resource
RidsNrrLASRohrer Resource
RidsNrrPMDavisBesse Resource
RidsRgn3MailCenter Resource
RidsNrrDeEvib Resource
RidsNrrDssSnpb Resource
SSheng, NRR
APatel, NRR

ADAMS Accession No. ML17256B190 ***by memo**

OFFICE	DORL/LPL3/PM	DORL/LPL3/LA	DE/EVIB/BC	DSS/SNPB/BC
NAME	BPurnell	SRohrer	DRudland*	RLukes*
DATE	10/04/17	09/19/17	07/25/17	07/25/17
OFFICE	OGC /NLO	DORL/LPL3/BC	DORL/LPL3/PM	
NAME	CKanatas	DWrona	BPurnell	
DATE	10/04/17	10/12/17	10/12/17	

OFFICIAL RECORD COPY