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L-15-186

10 CFR 54

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Davis-Besse Nuclear Power Station, Unit No. 1
Docket No. 50-346, License Number NPF-3
Supplemental Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, Reactor Vessel Internals Inspection Plan (TAC No. ME4640) and License Renewal Application Amendment No. 57

By letter dated August 27, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102450565), FirstEnergy Nuclear Operating Company (FENOC) 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, Unit No. 1 (Davis-Besse). By letter dated April 21, 2015 (ML15113B132, ML15113B133 and ML15113B134), FENOC submitted the Davis-Besse Reactor Vessel Internals Inspection Plan. By letter dated May 20, 2015, FENOC submitted supplemental information for the Inspection Plan. Based on a telephone conference call held with the Nuclear Regulatory Commission (NRC) staff on May 19, 2015, to discuss the Inspection Plan details, attached is supplemental information to support completion of the NRC staff review of the Plan.

The Attachment provides a description of the Davis-Besse Reactor Vessel Internals Inspection Plan supplemental information. The Enclosure provides Amendment No. 57 to the Davis-Besse LRA.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Clifford I. Custer, Fleet License Renewal Project Manager, at 724-682-7139.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 5th, 2015.

Sincerely,

Raymond A. Lieb

Raymond A. Lieb

Attachment:
Davis-Besse Reactor Vessel Internals (RVI) Inspection Plan Supplemental Information

Enclosure:
Amendment No. 57 to the Davis-Besse License Renewal Application

cc: NRC DLR Project Manager
NRC Region III Administrator

cc: w/o Attachment or Enclosure
NRC DLR Director
NRR DORL Project Manager
NRC Resident Inspector
Utility Radiological Safety Board

Item 1 – RVI Core Clamping Measurements:

By letter dated April 21, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML15113B132, ML15113B133 and ML15113B134), FENOC provided the Davis-Besse License Renewal Reactor Vessel Internals (RVI) Inspection Plan. The RVI Inspection Plan is included in the letter as Enclosure A, AREVA NP Licensing Report No. ANP-3290 Revision 1, "Reactor Vessel Internals Inspection Plan for the Davis-Besse Nuclear Power Plant Unit No. 1" (ML15113B132). ANP-3290, Section 4.2.8, "Core Clamping Measurements," stated that the Davis-Besse core clamping measurements were performed in 2014 during the Cycle 18 Refueling Outage. Core clamping measurements determine the difference in height between the minimum height of the vessel seating surface and the maximum thickness of the top of the plenum rib pads and the core support shield assembly (i.e., clamped components) to determine whether wear of the clamped components is occurring. According to MRP-227A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," the measured differential height from the top of the plenum rib pads to the vessel seating surface shall average less than 0.004 inches compared to the as-built condition.

During the Cycle 18 Refueling Outage, eight (8) core clamping measurements were obtained from the four cardinal directions and 45 degrees between. The average difference of the readings compared to as-built readings was less than 0.004 inches. The measured readings were uniform and no wear was apparent.

Item 2 – License Renewal Application (LRA) Table 3.1.2-2 Notes:

License Renewal Application (LRA) Table 3.1.2-2, "Aging Management Review Results – Reactor Vessel Internals," is revised to add a new plant-specific note to row number 11 (CSS Vent Valve Top Retaining Ring and Bottom Retaining Ring (primary component with no expansion components)) to credit Technical Specification 5.5.4, "Reactor Vessel Internals Vent Valves Program," for performance monitoring of the vent valves. The new note reads as follows:

> *Performance monitoring of CSS Vent Valve passive and long-lived components that are subject to aging management review (AMR) is conducted in accordance with Technical Specification 5.5.4, "Reactor Vessel Internals Vent Valves Program."*

Item 3 – RVI Replacement Bolts:

Section 3.5.1, subsection 5, of the NRC safety evaluation (Revision 1) for MPR-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," states the following:

> *For those cumulative usage factor (CUF) analyses that are TLAAs [Time-Limited Aging Analyses], the applicant may use the PWR Vessel Internals Program as the basis for accepting these CUF analyses in accordance with 10 CFR 54.21(c)(1)(iii) only if the RVI components in the CUF analyses are periodically inspected for fatigue-induced cracking in the components during the period of extended operation. The periodicity of the inspections of these components shall be justified to be adequate to resolve the TLAA.*

Locations using replacement bolts are the upper core barrel (UCB), the lower core barrel (LCB), lower thermal shield (LTS) and surveillance specimen holder tube (SSHT). The fatigue TLAA for these replacement bolts will be managed by the Fatigue Monitoring Program, and also the PWR Reactor Vessel Internals Program where volumetric UT examinations will be performed on a periodic basis consistent with the program's inspection plan, in accordance with 10 CFR 54.21(c)(1)(iii).

LRA Sections 4.3.2.2.2.1, "Low Cycle Fatigue," and A.2.3.2.1, "Reactor Vessel Internals Bolts," are revised to also credit the PWR Vessel Internals Program for managing the replacement bolting fatigue TLAA.

See the Enclosure to this letter for the revision to the Davis-Besse LRA.

Enclosure

Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse)

Letter L-15-186

Amendment No. 57 to the Davis-Besse License Renewal Application

License Renewal Application Sections Affected

Table 3.1.2-2

Table 3.1.2 Plant-Specific Notes

Section 4.3.2.2.2.1

Section A.2.3.2.1

The Enclosure identifies the change to the License Renewal Application (LRA) by Affected LRA Section, LRA Page No., and Affected Paragraph and Sentence. The count for the affected paragraph, sentence, bullet, etc. starts at the beginning of the affected Section or at the top of the affected page, as appropriate. Below each section the reason for the change is identified, and the sentence affected is printed in *italics* with deleted text *~~lined-out~~* and added text *underlined*.

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 3.1.2-2	**Page 3.1-97**	**Row 11**
Table 3.1.2 Plant-Specific Notes	**Page 3.1-187**	**1 New Note**

As a supplement to the Davis Besse Reactor Vessel Internals Inspection Plan submitted by FENOC letter dated April 21, 2015 (ML15113B132, ML15113B133 and ML15113B134), row number 11 of Table 3.1.2-2, "Aging Management Review Results – Reactor Vessel Internals," and Table 3.1.2 Plant Specific Notes, previously revised in FENOC Letters dated September 16, 2011 (ML11264A059) and March 9, 2012 (ML12094A383), are revised to read as follows:

Table 3.1.2-2 Aging Management Review Results – Reactor Vessel Internals

Row No.	Component Type	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Rev. 1 Item (Rev. 2 Item)	Table 1 Item	Notes
Core Support Shield (CSS) Assembly									
11	CSS Vent Valve Top Retaining Ring and Bottom Retaining Ring (primary component with no expansion components)	Support	Stainless Steel	Borated Reactor Coolant with Neutron Fluence	Reduction in fracture toughness	PWR Reactor Vessel Internals	IV.B4-16 (IV.B4.RP-252)	3.1.1-22	A *0117*

Plant-Specific Notes:	
0117	*Performance monitoring of CSS Vent Valve passive and long-lived components that are subject to aging management review (AMR) is conducted in accordance with Technical Specification 5.5.4, "Reactor Vessel Internals Vent Valves Program."*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
4.3.2.2.2.1	**Pages 4.3-8 & 4.3-9**	**3rd Paragraph, and Disposition**

As a supplement to the Davis-Besse Reactor Vessel Internals Inspection Plan, LRA Section 4.3.2.2.2.1, "Low Cycle Fatigue," is revised to read as follows:

The core support components are designed to meet the stress requirements of the ASME Section III during normal operation and transients. USAR Appendix 4A contains a detailed stress analysis of the internals under accident conditions. USAR Table 4.2-5 shows that stresses are within established limits, and that deflections would not prevent control rod assembly insertion.

Although the reactor vessel internals are designed to meet the stress requirements of ASME Section III, they are not code components. Consequently, a fatigue analysis of the reactor vessel internals was not performed as part of the original design. The stresses for faulted conditions were analyzed, but fatigue for normal and upset conditions was not analyzed.

Davis-Besse has replaced the majority of the Alloy A-286 bolts for the reactor vessel internals with Alloy X-750 HTH bolts. The replacement bolts were designed to ASME Section III, and fatigue analyses were performed for the replacement bolts. Davis-Besse has not replaced the upper thermal shield bolts, flow distributor bolts, or guide block bolts. All cumulative usage factors calculated for the reactor vessel internals bolts are based on the nuclear steam supply system design transients identified in Table 4.3-1, and are less than 1.0. Therefore, the effects of fatigue will be adequately managed for the period of extended operation by the Fatigue Monitoring Program. *In addition, the fatigue TLAA for the replacement bolts will be managed by the PWR Reactor Vessel Internals Program where volumetric UT examinations will be performed on a periodic basis consistent with the program's inspection plan.*

Disposition:	10 CFR 54.21(c)(1)(iii)	~~*The low-cycle fatigue analysis TLAA for the reactor vessel internals will be managed by the Fatigue Monitoring Program for the period of extended operation.*~~ *The effects of fatigue on the reactor vessel internals bolts will be managed by the Fatigue Monitoring Program and the PWR Reactor Vessel Internals Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).*

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
A.2.3.2.1	**Page A-38**	**3rd Paragraph**

As a supplement to the Davis-Besse Reactor Vessel Internals Inspection Plan, LRA Section A.2.3.2.1, "Reactor Vessel Internals Bolts," is revised to read as follows:

Although the reactor vessel internals are designed to meet the stress requirements of ASME Section III, they are not code components. Consequently, a fatigue analysis of the reactor vessel internals was not required and was not performed as part of the original design.

FENOC has replaced the majority of the stainless steel, Alloy A-286, bolts for the reactor vessel internals with Alloy X-750 HTH bolts at Davis-Besse. The replacement bolts were designed to ASME Section III, and are provided with fatigue analyses. FENOC has not replaced the upper thermal shield bolts, flow distributor bolts, or guide block bolts at Davis-Besse. Design cumulative usage factors for the reactor vessel internals bolts are based on design cycles.

The effects of fatigue on the reactor vessel internals bolts will be managed by the Fatigue Monitoring Program and the PWR Reactor Vessel Internals Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).