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November 2, 2012
L-12-406

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 Reply to Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1, License Renewal Application (TAC No. ME4640) and License Renewal Application Amendment No. 35

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). During a telephone conference call on October 23, 2012, the Nuclear Regulatory Commission (NRC) requested clarification regarding the response to NRC request for additional information (RAI) 4.2.4-1 related to pressure-temperature limits provided by FENOC letter dated August 24, 2012 (ML12240A219). Also, during a telephone conference call on October 16, 2012, the NRC requested clarification regarding a portion of the response to NRC RAI B.2.4-1 related to high strength structural bolting provided by FENOC letter dated May 24, 2011 (ML11151A090).

The Attachment provides the FENOC supplemental responses to the NRC requests for additional information. The NRC request is shown in bold text followed by the FENOC response. The Enclosure provides Amendment No. 35 to the Davis-Besse LRA.

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NRR

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 November 2, 2012.

Sincerely,

Raymond A. Lieb

Raymond A. Lieb

Attachment:
Supplemental Reply to Requests for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse), License Renewal Application, Sections 4.2.4 and B.2.4

Enclosure:
Amendment No. 35 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

Section 4.2.4

Supplemental Question RAI 4.2.4-1

The NRC initiated a telephone conference call with FENOC on October 23, 2012, to discuss the FENOC response to NRC request for additional information (RAI) 4.2.4-1 submitted by FENOC letter dated August 24, 2012 (ML12240A219).

NRC stated that they could not find information in the FENOC response to RAI 4.2.4-1 that addressed how future pressure-temperature limit curves would be developed for the period of extended operation taking into account the neutron embrittlement effects on the extended beltline region and the localized stresses of the inlet and outlet nozzles.

FENOC stated that the Davis-Besse pressure-temperature limit curves are currently limited to 32 effective full power years, and that additional analysis is required to extend the curves in the future.

Following discussions, both parties agreed that FENOC would submit a supplemental response to RAI 4.2.4-1 to clarify the response and incorporate the clarification into License Renewal Application (LRA) Sections 4.2.4 and A.2.2.4, both titled "Pressure-Temperature Limits."

SUPPLEMENTAL RESPONSE RAI 4.2.4-1

BAW-10046A, Revision 2 [Reference 1], concludes that the reactor vessel closure head region (subjected to significant stresses due to mechanical loads resulting from bolt preload), the reactor vessel outlet nozzles (inside corner of the nozzle is subjected to high local stresses produced by pressure), and the beltline region are the only portions of the reactor coolant pressure boundary that, at different stages of the vessel's design life, regulate the pressure-temperature limitations for normal operation and inservice pressure tests.

The beltline or beltline region of reactor vessel is defined by 10 CFR 50 Appendix G Section II.F, as the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

As listed in LRA Section 4.2.1.3, "Beltline Evaluation," the beltline materials at 40 years for Davis-Besse include the following items:

- Nozzle Belt Forging (ADB 203)
- Upper Shell Forging (AKJ 233)
- Lower Shell Forging (BCC 241)
- Nozzle Belt Forging to Upper Shell Forging Circumferential Weld (Inside 9%) (WF-232) / (Outside 91%) (WF-233)
- Upper Shell Forging to Lower Shell Forging Circumferential Weld (WF-182-1)

The Davis-Besse pressure-temperature limits reported in Reference 2, valid to 32 Effective Full Power Years (EFPY) of operation or April 22, 2017, whichever occurs first, are based on evaluation of the 40-year beltline materials listed above, the reactor vessel closure head region and the reactor vessel outlet nozzles.

As provided in Section 4.2.1.3 of the LRA, the beltline materials for the period of extended operation include all items with 52 EFPY inside surface fluence greater than 1.0E+17 n/cm^2. For Davis-Besse, the 60-year beltline items include the 40-year items listed above plus the following items.

- Reactor Vessel Inlet Nozzle Forgings (BSS 270)
- Reactor Vessel Outlet Nozzle Forgings (ATS 239)
- Dutchman Forging (122Y384VA1)
- Nozzle Belt Forging to Bottom of Reactor Vessel Inlet Nozzle Forging Weld (WF-233 / WF-232)
- Nozzle Belt Forging to Bottom of Reactor Vessel Outlet Nozzle Forging Weld (WF-233)
- Lower Shell Forging to Dutchman Forging Circumferential Weld (Inside 12%) (WF-232) / (Outside 88%) (WF-233)

The revised pressure-temperature limits for the period of extended operation will be based on an evaluation of the effects of neutron embrittlement for the 60-year beltline materials, the stresses in the closure head region of the reactor vessel (subject to significant stresses due mechanical loads resulting from bolt preload) and the stresses in the reactor vessel outlet nozzles (largest nozzles in the Reactor Coolant System and the inside corners of the nozzles are subjected to high local stresses produced by pressure).

LRA Sections 4.2.4 and A.2.2.4 are revised consistent with this response.

References for this response:

1) AREVA NP Document BAW-10046A, Revision 2 "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50 Appendix G," June 1986

2) Davis-Besse Nuclear Power Station, Unit No.1, Docket No. 50-346, License No. NPF-3, Pressure and Temperature Limits Report (ML11304A188)

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

Section B.2.4

Supplemental Question RAI B.2.4-1

The NRC initiated a telephone conference call with FENOC on October 16, 2012, to request clarification of the FENOC response regarding the management of high strength bolting. Following discussions, both parties agreed that the following items should be addressed in a FENOC supplemental response to request for additional information (RAI) B.2.4-1:

- **Provide clarification regarding the initial high strength bolting response to RAI B.2.4-1, specifically addressing statements regarding inspection of high strength bolts.**
- **Include a discussion regarding the use of molybdenum disulfide (MoS_2) as a lubricant on high strength bolting.**

SUPPLEMENTAL RESPONSE RAI B.2.4-1

In response to RAI B.2.4-1, FENOC provided the following discussion in FENOC letter dated May 24, 2011 (ML11151A090) (Attachment A, page 16 of 44):

> Detection of aging effects:
>
> Structural bolting, including component support bolting, both inside and outside containment, is inspected by visual inspection through the Inservice Inspection (ISI) Program – IWF and Structures Monitoring Program. Containment penetration pressure retaining bolting is inspected by visual inspection through the ISI Program – IWE. If any degradation of these bolts and fasteners is identified, a closer inspection is performed to

assess the extent of degradation. An appropriate technique (i.e., visual inspection or volumetric examination) is selected on the basis of the bolting application and the applicable code.

Structural bolting materials used at Davis-Besse include A 36, A 276, A 307, A 325, A 449, A 490, and A 540, conforming to ASTM standards. Volumetric or surface examinations are not currently conducted for stress corrosion cracking susceptible bolts since no instances of failed bolting or bolted connections due to stress corrosion cracking had occurred at Davis-Besse. For stress corrosion cracking to occur in a susceptible high strength bolting material, a sustained tensile stress and a corrosive environment must be present. Visual examinations of structural assemblies will detect corrosion or conditions indicative of a corrosive environment that could lead to stress corrosion cracking in potentially susceptible high strength bolting, and will cause appropriate corrective action to be taken under the Corrective Action Program when necessary. Corrective action may include volumetric examination of affected bolts, hammer testing, or other actions appropriate for the condition. Therefore, visual examination, as described, will effectively manage the aging of installed structural high strength bolting.

LRA Table 3.5.2-13, "Aging Management Review Results - Bulk Commodities," rows 138, 140, 146, 149, 158, and 162 are consistent with NUREG-1801,Rev. 1, Volume 2 line item III.B.1.1-3, where cracking of anchor bolts is managed by the XI.M18, "Bolting Integrity," program. LRA Table 3.5.2-13 is revised to include a plant-specific note to clarify that the Bolting Integrity Program includes the Inservice Inspection (ISI) Program – IWE, Inservice Inspection (ISI) Program – IWF, and Structures Monitoring Program for the management of structural bolting.

To provide clarification on testing and lubrication practices of high strength structural bolts, the second paragraph of "Detection of aging effects" (above) is replaced in its entirety to read as follows:

Structural bolting materials used at Davis-Besse include A 36, A 276, A 307, A 325, A 449, A 490, and A 540, conforming to ASTM standards. The specified high-strength bolts used for structural steel at Davis-Besse are constructed of the A 325 or A 490 material. Table 3.5-1, "Summary of Aging Management Programs for Containments, Structures and Supports Evaluated in Chapters II and III," item 3.5.1-69 of NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," Revision 2, states that, "ASTM A 325, F1852, and ASTM A 490 bolts used in civil structures have not shown to be prone to SCC. SCC potential need not be evaluated for these bolts." In addition, use of molybdenum disulfide (MoS_2) as a lubricant has been shown to be a potential contributor to stress corrosion cracking (SCC) and should not be used. Lubrication is not applied to the threads of structural bolting at Davis-Besse, unless otherwise specified. There is no lubricant specified or used for the A 325 and A 490 high strength structural bolts at Davis-Besse. Therefore, visual examination, as described, will effectively manage the aging of installed structural high strength bolting.

Enclosure

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

Letter L-12-406

Amendment No. 35 to the
Davis-Besse License Renewal Application

License Renewal Application
Sections Affected

Section 4.2.4

Section A.2.2.4

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**
4.2.4	**Page 4.2-11**	**5th Paragraph**

Based on the supplemental response to RAI 4.2.4-1, the 5th paragraph of LRA Section 4.2.4, "Pressure-Temperature Limits," previously revised in FENOC letter dated August 24, 2012 (ML12240A219), is revised to read as follows:

The current P-T limits, generated consistent with the requirements of 10 CFR 50 Appendix G and Regulatory Guide 1.99 Revision 2, are valid until 32 EFPY, or April 22, 2017, whichever occurs first. A revised pressure and temperature limits report will be submitted to the NRC, in accordance with Technical Specification 5.6.4, before Davis-Besse operates beyond 32 EFPY, or April 22, 2017, whichever occurs first, in accordance with the requirements of 10 CFR 50, Appendix G. ~~*The P-T limit curves, as contained in the pressure-temperature limit report and providing the information required by Technical Specification 5.6.4, will be updated as necessary through the period of extended operation as part of the Reactor Vessel Surveillance Program.*~~ *The revised P-T limits for the period of extended operation will be based on an evaluation of the effects of neutron embrittlement for the 60-year beltline materials, the stresses in the closure head region of the reactor vessel (subject to significant stresses due mechanical loads resulting from bolt preload) and the stresses in the reactor vessel outlet nozzles (largest nozzles in the RCS and the inside corners of the nozzles are subjected to high local stresses produced by pressure). The 60-year reactor vessel beltline materials are listed as follows:*

- *Nozzle Belt Forging (ADB 203)*
- *Upper Shell Forging (AKJ 233)*
- *Lower Shell Forging (BCC 241)*
- *Nozzle Belt Forging to Upper Shell Forging Circumferential Weld (Inside 9%) (WF-232) / (Outside 91%) (WF-233)*
- *Upper Shell Forging to Lower Shell Forging Circumferential Weld (WF-182-1)*
- *Reactor Vessel Inlet Nozzle Forgings (BSS 270)*
- *Reactor Vessel Outlet Nozzle Forgings (ATS 239)*
- *Dutchman Forging (122Y384VA1)*

- *Nozzle Belt Forging to Bottom of Reactor Vessel Inlet Nozzle Forging Weld (WF-233 / WF-232)*
- *Nozzle Belt Forging to Bottom of Reactor Vessel Outlet Nozzle Forging Weld (WF-233)*
- *Lower Shell Forging to Dutchman Forging Circumferential Weld (Inside 12%) (WF-232) / (Outside 88%) (WF-233)*

Revisions to the P-T limits will be managed as part of the Reactor Vessel Surveillance Program for the period of extended operation.

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
A.2.2.4	**Page A-33**	**3rd Paragraph**

Based on the supplemental response to RAI 4.2.4-1, the 3rd paragraph of LRA Section A.2.2.4, "Pressure-Temperature Limits," previously revised in FENOC letter dated August 24, 2012 (ML12240A219), is revised to read as follows:

The current P-T limits, generated consistent with the requirements of 10 CFR 50 Appendix G and Regulatory Guide 1.99 Revision 2, are valid until 32 EFPY, or April 22, 2017, whichever occurs first. A revised pressure and temperature limits report (PTLR) will be submitted to the NRC, in accordance with Technical Specification 5.6.4, before Davis-Besse operates beyond 32 EFPY, or April 22, 2017, whichever occurs first, in accordance with the requirements of 10 CFR 50, Appendix G. ~~*The P-T limit curves, as contained in the PTLR, will be updated as necessary through the period of extended operation as part of the Reactor Vessel Surveillance Program.*~~ *The revised P-T limits for the period of extended operation will be based on an evaluation of the effects of neutron embrittlement for the 60-year beltline materials, the stresses in the closure head region of the reactor vessel (subject to significant stresses due mechanical loads resulting from bolt preload) and the stresses in the reactor vessel outlet nozzles (largest nozzles in the RCS and the inside corners of the nozzles are subjected to high local stresses produced by pressure). The 60-year reactor vessel beltline materials are listed as follows:*

- *Nozzle Belt Forging (ADB 203)*
- *Upper Shell Forging (AKJ 233)*
- *Lower Shell Forging (BCC 241)*
- *Nozzle Belt Forging to Upper Shell Forging Circumferential Weld (Inside 9%) (WF-232) / (Outside 91%) (WF-233)*
- *Upper Shell Forging to Lower Shell Forging Circumferential Weld (WF-182-1)*
- *Reactor Vessel Inlet Nozzle Forgings (BSS 270)*
- *Reactor Vessel Outlet Nozzle Forgings (ATS 239)*
- *Dutchman Forging (122Y384VA1)*
- *Nozzle Belt Forging to Bottom of Reactor Vessel Inlet Nozzle Forging Weld (WF-233 / WF-232)*

- *<u>Nozzle Belt Forging to Bottom of Reactor Vessel Outlet Nozzle Forging Weld (WF-233)</u>*
- *<u>Lower Shell Forging to Dutchman Forging Circumferential Weld (Inside 12%) (WF-232) / (Outside 88%) (WF-233)</u>*