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August 24, 2012
L-12-324

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
Reply to Supplemental 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. 33

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 May 31, 2012 (ML12144A038), the Nuclear Regulatory Commission (NRC) requested additional information to complete its review of the License Renewal Application (LRA). During a telephone conference call held on June 21, 2012, the NRC informed FENOC that a supplemental request for additional information (RAI) would be issued to replace the RAI in NRC letter dated May 31, 2012 (ML12144A038). By letter dated July 26, 2012 (ML12201B519), the NRC refined and reiterated the request as a supplemental RAI.

The Attachment provides the FENOC reply to the NRC supplemental RAI. The NRC request is shown in bold text followed by the FENOC response. The Enclosure provides Amendment No. 33 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 August 24, 2012.

Sincerely,

Barry S. Allen

Barry S. Allen

Attachment:
Reply to Supplemental Request for Additional Information for the Review of the Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse), License Renewal Application (LRA), Section 4.2.4

Enclosure:
Amendment No. 33 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

<u>Section 4.2.4</u>

Question RAI 4.2.4-1 – Pressure-Temperature (P-T) Limits

<u>Background</u>:

The Davis-Besse License Renewal Application (LRA), Section 4.2.4 describes the time-limited aging analysis for the pressure-temperature (P-T) limit curves at Davis-Besse. As stated in LRA Section 4.2.4, the Davis-Besse P-T limit curves are established in a P-T Limits Report (PTLR), the contents of which are controlled in accordance with Technical Specification (TS) 5.6.4 requirements. The current Davis-Besse PTLR contains P-T limit curves that are valid through 32 effective full power years of facility operation. LRA Section 4.2.4 states that the P-T limit curves, as established in the PTLR, will be updated as necessary in accordance with TS 5.6.4 requirements and managed for the period of extended operation, as part of the Reactor Vessel Surveillance Program (LRA Appendix B, Section B.2.35), in accordance with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 54.21 (c)(1)(iii).

10 CFR Part 50, Appendix G, Paragraph IV.A states that, "*the pressure-retaining components of the reactor coolant pressure boundary [RCPB] that are made of ferritic materials must meet the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code [ASME Code, Section III], supplemented by the additional requirements set forth in* [paragraph IV.A.2, "Pressure-Temperature (P-T) Limits and Minimum Temperature Requirements"]..." Therefore, 10 CFR Part 50, Appendix G requires that P-T limits be developed for the ferritic materials in the reactor vessel (RV) beltline (neutron fluence ≥ $1x10^{17}$ n/cm^2, E > 1 MeV), as well as ferritic materials not in the RV beltline (neutron fluence < $1x10^{17}$ n/cm^2, E > 1 MeV). Further, 10 CFR Part 50, Appendix G requires that all RCPB components must meet the American Society of Mechanical Engineers (ASME) Code, Section III requirements. The relevant ASME Code, Section III requirement that will affect the P-T limits is the lowest service temperature requirement for all RCPB components specified in Section III, NB-2332(b).

<u>Issue</u>:

P-T limit calculations for ferritic RCPB components that are not RV beltline shell materials may define P-T curves that are more limiting than those calculated for the RV beltline shell materials. This may be due to the following factors:

1. **RV nozzles, penetrations, and other discontinuities have complex geometries that may exhibit significantly higher stresses than those for the RV beltline shell region. These higher stresses can potentially result in more restrictive P-T limits, even if the reference temperature (RT_{NDT}) for these components is not as high as that of RV beltline shell materials that have simpler geometries.**

2. **Ferritic RCPB components that are not part of the RV may have initial RT_{NDT} values, which may define a more restrictive lowest operating temperature in the P-T limits than those for the RV beltline shell materials.**

<u>Request</u>:

Describe how the P-T limit curves to be developed for use in the period of extended operation, and the methodology used to develop these curves, consider all RV materials (beltline and nonbeltline) and the lowest service temperature of all ferritic RCPB materials, consistent with the requirements of 10 CFR Part 50, Appendix G.

RESPONSE RAI 4.2.4-1 – Pressure-Temperature (P-T) Limits

FirstEnergy Nuclear Operating Company (FENOC) used the methods described in Topical Report BAW-10046A, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50 Appendix G," Revision 2 [Reference 1], to develop the pressure-temperature (P-T) limits for Davis-Besse. For Babcock & Wilcox (B&W) nuclear steam systems, Topical Report BAW-10046A describes methods for compliance with the requirements of 10 CFR 50 Appendix G, "Fracture Toughness Requirements." This report addresses all reactor coolant pressure boundary components (beltline and non-beltline). The NRC has reviewed the methods described in Topical Report BAW-10046A and approved the report by issuance of a Safety Evaluation Report (SER) dated April 30, 1986 [Reference 2]. Additional details for development of the Davis-Besse pressure and temperature (P-T) limits are provided below.

The Davis-Besse Pressure and Temperature Limits Report (PTLR) [Reference 3] provides P-T limits for the Davis-Besse reactor coolant pressure boundary (RCPB) that are valid to 32 Effective Full Power Years (EFPY) of operation or April 22, 2017, whichever occurs first. The P-T limits were generated consistent with the requirements of 10 CFR 50 Appendix G and Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, using the methods described in Topical Report BAW-10046A, Revision 2, and ASME Section XI, Appendix G, as modified by the alternative rules provided in ASME Code Case N-588 for flaws in circumferential welds and ASME Code Case N-640 for use of the K_{IC} reference fracture toughness curve from Section XI, Appendix A. The reference temperature for nil-ductility transition (RT_{NDT}) values of the reactor vessel beltline welds (Linde 80 welds) were determined using methods provided in approved Topical Report BAW-2308, "Initial RT_{NDT} Of Linde 80 Weld Materials," Revisions 1-A [Reference 4] and 2-A [Reference 5], rather than the methodology described within Topical Report BAW-10046A, Revision 2, which is used to evaluate the other beltline components. The NRC required licensees to obtain an exemption from 10 CFR 50.61 and 10 CFR 50, Appendix G to use the alternate initial RT_{NDT} values provided in BAW-2308, Revisions 1-A and 2-A. The required exemption was granted by the NRC by letter dated December 14, 2010 (ML103060213) [Reference 6].

BAW-10046A, Revision 2, is applicable to all current B&W nuclear steam systems. Section 2 of BAW-10046A includes a list of all RCPB items used in the fabrication of B&W-designed plant components. Section 3 describes the material properties (including initial RT_{NDT} and C_vUSE (upper shelf energy)) of these ferritic RCPB items. Considering all ferritic RCPB items and consideration of lowest service temperatures, Section 4, Pages 4-1 and 4-2 of BAW-10046A, concludes that the reactor vessel closure head region, the reactor vessel outlet nozzles, and the beltline region are the only portions of the RCPB that, at different stages of the vessel's design life, regulate the pressure-temperature limitations for normal operation and inservice pressure tests.

For beltline materials, AREVA NP Inc. (AREVA) has traditionally selected the reactor vessel (RV) material with the highest adjusted reference temperature as limiting for the evaluation of P-T limits. The controlling beltline material for Davis-Besse has traditionally been the upper shell to lower shell circumferential seam weld WF-182-1. AREVA recently investigated the impact of stress intensity on P-T limits for Davis-Besse by evaluating the upper transition weld WF-232/233, which is directly below a taper transition. The evaluation concluded that the stress intensity factors for weld WF-232/233 exceed the stress intensity factors for weld WF-182-1; however, the material properties at weld WF-232/233 are significantly better than weld WF-182-1, and more than compensates for the higher stress intensity factors. Therefore, weld WF-182-1 continues to be the limiting beltline material relative to establishment of P-T limits for Davis-Besse.

As described in BAW-10046A, non RV beltline materials have always been considered when establishing P-T limits for Davis-Besse. Page 4-2 of BAW-10046A reports that the outlet nozzle of the reactor vessel is the largest nozzle in the Reactor Coolant System (RCS) and the inside corner of the nozzle is subjected to high local stresses produced by pressure. The RV outlet nozzle, due to consideration of loading conditions, is more limiting relative to stress than any of the Class 1 ferritic branch connections (e.g., hot leg surge nozzle) and the large bore RCS piping and the primary nozzles of the steam generators.

With regard to replacement of Class 1 ferritic RCPB items (e.g., RV closure head and any future replacement of RCPB components), ASME III NB-3211(d) requires that protection against nonductile fracture be provided by satisfying one of the following provisions:

1. performing an evaluation of service and test conditions by methods similar to those contained in Appendix G; or

2. for piping, pump, and valve material thickness greater than 2½ in. (64 mm) establishing a lowest service temperature that is not lower than RT_{NDT} (NB-2331) + 100°F (56°C);

3. for piping, pump, and valve material thickness equal to or less than 2½ in. (64 mm), the requirements of NB-2332(a) shall be met at or below the lowest service temperature as established in the design specification.

Therefore, for replacement components, an ASME Section III, Appendix G, analysis is required to ensure that the new component is bounded by the ASME Section XI, Appendix G, analysis of the RV used to derive the P-T limits.

LRA Sections 4.2.4 and A.2.2.4, both titled "Pressure-Temperature Limits," are revised consistent with this response and to show that the current P-T limits are valid to 32 EFPY or April 22, 2017 (expiration of the current facility operating license), whichever occurs first. Also, LRA Sections 4.2.5 and A.2.2.5, both titled "Low-Temperature Overpressure Protection Limits," are revised to show that the current technical specifications for low-temperature overpressure protection (LTOP) are valid to 32 EFPY. In addition, LRA Sections 4.8, "References," and A.2.8, "Appendix A.2 References," are revised to add new references.

References

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) NRC Letter, Dennis M. Crutchfield (NRC), to James H. Taylor (Babcock and Wilcox Company), Acceptance for Referencing of Licensing Topical Report BAW-10046, Rev. 2 B&W Owners Group Materials Committee "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," April 30, 1986

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

4) AREVA NP Document BAW-2308 Revision 1-A, "Initial RT_{NDT} Of Linde 80 Weld Materials," August 2005

5) AREVA NP Document BAW-2308 Revision 2-A, "Initial RT_{NDT} Of Linde 80 Weld Materials," March 2008

6) NRC Letter to FirstEnergy Nuclear Operating Company, "Davis-Besse Nuclear Power Station, Unit 1-Exemption from the Requirements of 10 CFR Part 50.61 and 10 CFR Part 50, Appendix G," dated December 14, 2010 (ML103060213)

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

Correction to the Davis-Besse License Renewal Application

In August 2012, it was discovered that an error was made in the Davis-Besse adjusted reference temperature (ART) calculations for 52 EFPY. Incorrect dimensions were used for the thickness of the Dutchman forging and the lower shell to Dutchman weld (OD 88%) that resulted in incorrect ART values at ¼T and ¾T for these components in LRA Table 4.2-4 "ARTs at 52 EFPY for Davis-Besse Reactor Vessel Beltline Materials." The Dutchman forging and the lower shell to Dutchman weld (OD 88%) were not 40-year beltline materials, and therefore, the current PTLR (Revision 1) is not affected.

LRA Table 4.2-4 is revised to show the corrected ART values including the corrected inputs to the ART calculations for the Dutchman forging and the lower shell to Dutchman weld (OD 88%). Also, LRA Section 4.2.4 is revised to include the thickness values for the subject weld and forging.

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-12-324

Amendment No. 33 to the Davis-Besse License Renewal Application

License Renewal Application Sections Affected

Section 4.2.4

Table 4.2-4

Section 4.2.5

Section 4.8

Section A.2.2.4

Section A.2.2.5

Section A.2.8

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**	**Entire section**

In response to RAI 4.2.4-1, LRA Section 4.2.4, "Pressure-Temperature Limits," is revised to read as follows:

4.2.4 PRESSURE-TEMPERATURE LIMITS

10 CFR 50 Appendix G requires the establishment of pressure and temperature (P-T) limits for material fracture toughness requirements of the reactor coolant pressure boundary materials. Appendix G mandates the use of the ASME Section III, Appendix G to determine the stresses and fracture toughness at locations within the reactor coolant pressure boundary.

The Davis-Besse Pressure and Temperature Limits Report (PTLR) [Reference 4.8-21] provides pressure-temperature (P-T) limits for the Davis-Besse reactor coolant pressure boundary (RCPB) that are valid to 32 Effective Full Power Years (EFPY) of operation or April 22, 2017, whichever occurs first. The P-T limits were generated consistent with the requirements of 10 CFR 50 Appendix G and Regulatory Guide 1.99, Revision 2, using the methods described in Topical Report BAW-10046A, Revision 2 [Reference 4.8-8], and ASME Section XI, Appendix G, as modified by the alternative rules provided in ASME Code Case N-588 for flaws in circumferential welds and ASME Code Case N-640 for use of the K_{IC} reference fracture toughness curve from Section XI, Appendix A. The reference temperature for nil-ductility transition (RT_{NDT}) values of the reactor vessel beltline welds (Linde 80 welds) were determined using methods provided in approved Topical Report BAW-2308, Revisions 1-A [Reference 4.8-14] and 2-A [Reference 4.8-22] rather than the methodology described within Topical Report BAW-10046A, Revision 2, which is used to evaluate the other beltline components. The NRC required licensees to obtain an exemption from 10 CFR 50.61 and 10 CFR 50, Appendix G to use the alternate initial RT_{NDT} values provided in BAW-2308 Revisions 1-A and 2-A. The required exemption was granted by the NRC in a letter dated December 14, 2010 [Reference 4.8-23].

One measure of the fracture toughness of a material is the reference temperature for nil-ductility transition (RT_{NDT}). *RT_{NDT} will increase with cumulative exposure to neutron irradiation resulting in an adjusted reference temperature (ART).* This ART is used in the development of P-T limit curves.

Table 4.2-4 includes 52 EFPY ART at the ¼T and ¾T locations for all 60-year beltline materials using Regulatory Guide 1.99, Revision 2, Position 1.1. *Minimum cladding thickness is 0.125 inches and the vessel low alloy steel* *thickness for the upper shell and lower shell forgings is 8.44 inches, 8.563 inches for the nozzle belt forging, 5.375 inches for the Dutchman forging, 5.375 inches for the lower shell forging to Dutchman forging weld, and 12.0 inches for the inlet and outlet nozzle forgings.* Using these vessel wall depths and the neutron fluence at the inner wetted surface of the vessel, the ¼T and ¾T fluence values for the Davis-Besse reactor vessel materials are calculated in accordance with Equation 1 of Regulatory Guide 1.99 Revision 2. Fluence values at the ¼T and ¾T locations for the RV inlet and outlet nozzle and associated welds that connect the nozzles to the nozzle belt forging were obtained by adding the attenuation from both the inside and outside surface. Position 2.1 was not used since two sets of credible ART surveillance data were not available. *Initial RT_{NDT} and margins for weld WF-182-1 and WF-233 are obtained from BAW-2308, Revision 1-A [Reference 4.8-14].*[3]

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 ~~21~~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 ~~21~~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.

Disposition: 10 CFR 54.21(c)(1)(iii) Reactor vessel P-T limits will be managed, as part of the Reactor Vessel Surveillance Program for the period of extended operation.

[3] ~~*FENOC submitted a request (FENOC Letter L-09-225 [Reference 4.8-16]) for exemption to use an alternate method, as described in approved topical report BAW-2308, Revision 1-A, for determining initial RT_{NDT} values of the Linde 80 weld materials present in the beltline region of the Davis-Besse reactor pressure vessel.*~~

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
Table 4.2-4	**Page 4.2-12**	**Dutchman Forging row, and Dutchman Forging Weld (OD 88%) row**

LRA Table 4.2-4, "ARTs at 52 EFPY for Davis-Besse Reactor Vessel Beltline Materials," is revised to correct adjusted reference temperature (ART) values, including the inputs to the ART calculations, for the Dutchman forging and the lower shell to Dutchman weld (OD 88%), to read as follows:

Table 4.2-4 ARTs at 52 EFPY for Davis-Besse Reactor Vessel Beltline Materials
(RG 1.99 Position 1.1)

Item	Material ID	ART, °F		Fluence n/cm^2 10E18		$RT_{NDT(u)}$, °F	ΔRT_{NDT}, °F		Fluence Factor		Chem. Factor	Margin, °F		Cu %	Ni %
		¼T	¾T	¼T	¾T		¼T	¾T	¼T	¾T		¼T	¾T		
Reactor Vessel Forgings															
Dutchman Forging SA-580 Class 2	122Y384VA1	~~76.1~~ 77.6	~~70.3~~ 73.0	~~0.136~~ 0.164	~~0.0495~~ 0.0859	3	~~10.3~~ 11.6	~~5.1~~ 7.5	~~0.135~~ 0.152	~~0.067~~ 0.099	76.1	~~62.8~~ 63.1	~~62.2~~ 62.5	0.11	0.74
Reactor Vessel Welds															
Lower Shell Forging to Dutchman Forging Circumferential Weld (OD 88%)	WF-233	~~63.9~~ 68.5	~~47.5~~ 55.0	~~0.136~~ 0.164	~~0.0495~~ 0.0859	-5	~~23.2~~ 26.2	~~11.5~~ 17.1	~~0.135~~ 0.152	~~0.067~~ 0.099	172.3	~~45.7~~ 47.3	~~41.0~~ 42.9	0.21	0.65

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
4.2.5	**Page 4.2-13**	**Third paragraph**

In response to RAI 4.2.4-1, the third paragraph of LRA Section 4.2.5, "Low-Temperature Overpressure Protection Limits," is revised to read:

The current technical specifications for LTOP are valid ~~through 21 EFPY~~ to 32 EFPY. These technical specifications used an improved methodology to calculate LTOP limits in accordance with generically approved topical report BAW-10046A [Reference 4.8-8].

Affected LRA Section	**LRA Page No.**	**Affected Paragraph and Sentence**
4.8	**Page 4.8-1 & 2**	**2 revised references; and, 3 new references**

In response to RAI 4.2.4-1, LRA Section 4.8, "References," previously revised in FENOC letters dated April 15, 2011 (ML11109A083) and August 17, 2011 (ML11231A966), is revised to correct the revision number of Topical Report BAW-10046A, correct the reference number of Report SIR-99-040 and add 3 new references:

Corrections:

4.8-8 *AREVA NP Document BAW-10046A, "Method of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G," Revision ~~4~~ 2*

4.8-~~18~~20 Structural Integrity Associates Report SIR-99-040, "ASME Code Case N-481, Evaluation of Davis-Besse Reactor Coolant Pumps" Rev. 1, September 2000 (ADAMS Accession No. ML011200090)

New references:

4.8-21 *Davis-Besse Nuclear Power Station, Unit No.1 ,Docket No. 50-346, License No. NPF-3, Pressure and Temperature Limits Report (ML11304A188)*

4.8-22 *AREVA NP Document BAW-2308 Revision 2-A, "Initial RT_{NDT} Of Linde 80 Weld Materials," March 2008 (NRC SER Included)*

4.8-23 *NRC Letter to FirstEnergy Nuclear Operating Company, "Davis-Besse Nuclear Power Station, Unit 1-Exemption from the Requirements of 10 CFR Part 50.61 and 10 CFR Part 50, Appendix G," dated December 14, 2010 (ML103060213)*

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.2.2.4	**Page A-33**	**Entire section**

In response to RAI 4.2.4-1, LRA Section A.2.2.4, "Pressure-Temperature Limits," is revised to read:

A.2.2.4 Pressure-Temperature Limits

10 CFR 50 Appendix G requires the establishment of pressure and temperature (P-T) limits for material fracture toughness requirements of the reactor coolant pressure boundary materials. Appendix G mandates the use of the ASME Section III, Appendix G to determine the stresses and fracture toughness at locations within the reactor coolant pressure boundary.

The Davis-Besse Pressure and Temperature Limits Report (PTLR) [Reference A.2-19] provides pressure-temperature (P-T) limits for the Davis-Besse reactor coolant pressure boundary (RCPB) that are valid to 32 Effective Full Power Years (EFPY) of operation or April 22, 2017, whichever occurs first. The P-T limits were generated consistent with the requirements of 10 CFR 50 Appendix G and Regulatory Guide 1.99, Revision 2 [Reference A.2-9], using the methods described in Topical Report BAW-10046A, Revision 2 [Reference A.2-16], and ASME Section XI, Appendix G, as modified by the alternative rules provided in ASME Code Case N-588 for flaws in circumferential welds and ASME Code Case N-640 for use of the K_{IC} *reference fracture toughness curve from Section XI, Appendix A. The reference temperature for nil-ductility transition (*RT_{NDT}*) values of the reactor vessel beltline welds (Linde 80 welds) were determined using methods provided in approved Topical Report BAW-2308, Revisions 1-A [Reference A.2-20] and 2-A [Reference A.2-21] rather than the methodology described within Topical Report BAW-10046A, Revision 2, which is used to evaluate the other beltline components. The NRC required licensees to obtain an exemption from 10 CFR 50.61 and 10 CFR 50, Appendix G to use the alternate initial* RT_{NDT} *values provided in BAW-2308 Revisions 1-A and 2-A. The required exemption was granted by the NRC in a letter dated December 14, 2010 [Reference A.2-22].*

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 ~~21~~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 ~~21~~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.

Reactor vessel P-T limits will be managed, as part of the Reactor Vessel Surveillance 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.2.2.5	**Page A-33**	**Third paragraph**

In response to RAI 4.2.4-1, the third paragraph of LRA Section A.2.2.5, "Low-Temperature Overpressure Protection Limits," is revised to read:

The current technical specifications for LTOP are valid ~~through 21 EFPY~~ to 32 EFPY. These technical specifications used an improved methodology to calculate LTOP limits in accordance with generically approved topical report BAW-10046A [Reference A.2-16]. Maintaining the LTOP limits in accordance with Appendix G of ASME Section XI, as required by Appendix G of 10 CFR 50, assures that the effects of aging on the intended functions will be adequately managed for the period of extended operation.

Affected LRA Section	LRA Page No.	Affected Paragraph and Sentence
A.2.8	**Page A-51 & 52**	**1 revised reference; and, 4 new references**

In response to RAI 4.2.4-1, LRA Section A.2.8, "Appendix A.2 References," previously revised by FENOC letter dated August 17, 2011 (ML11231A966), is revised to correct the revision number of Topical Report BAW-10046A and add 4 new references:

Corrections:

A.2-16 AREVA NP Document BAW-10046A, "Method of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G," Revision ~~4~~ 2

New references:

A.2-19 Davis-Besse Nuclear Power Station, Unit No.1 ,Docket No. 50-346, License No. NPF-3, Pressure and Temperature Limits Report (ML11304A188)

A.2-20 AREVA NP Document BAW-2308 Revision 1-A, "Initial RT_{NDT} *Of Linde 80 Weld Materials," March 2008*

A.2-21 AREVA NP Document BAW-2308 Revision 2-A, "Initial RT_{NDT} *Of Linde 80 Weld Materials," March 2008*

A.2-22 NRC Letter to FirstEnergy Nuclear Operating Company, "Davis-Besse Nuclear Power Station, Unit 1-Exemption from the Requirements of 10 CFR Part 50.61 and 10 CFR Part 50, Appendix G," dated December 14, 2010 (ML103060213)