FENOC™
FirstEnergy Nuclear Operating Company

5501 North State Route 2
Oak Harbor, Ohio 43449

Raymond A. Lieb
Vice President, Nuclear

419-321-7676
Fax: 419-321-7582

September 22, 2014
L-14-289

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 No. NPF-3
Pressure and Temperature Limits Report, Revision 2

In accordance with Technical Specification 5.6.4, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," FirstEnergy Nuclear Operating Company hereby submits Revision 2 of the PTLR for the Davis-Besse Nuclear Power Station, Unit No. 1, which was issued on September 15, 2014.

Revision 2 reissued the 32 Effective Full Power Years (EFPY) Pressure-Temperature limits to incorporate Revision 4 of ANP-2718, "Appendix G Pressure-Temperature Limits for 52 EFPY, Using ASME Code Cases for Davis-Besse Nuclear Power Station." Revision 4 of ANP-2718 combined the Heatup/Cooldown Curves into a single figure. This resulted in the re-numbering of the PTLR In-Service Leak and Hydrostatic Tests Figure from Figure 3 to Figure 2. No methodology changes occurred in this revision.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager - Fleet Licensing, at (330) 315-6810.

Sincerely,

Raymond A. Lieb

Enclosure:
FirstEnergy Nuclear Operating Company, Davis-Besse Unit 1, Pressure and Temperature Limits Report for the Earlier of 32 Effective Full Power Years or April 22, 2017, Revision 2

cc: NRC Region III Administrator
NRC Resident Inspector
NRC Project Manager
Utility Radiological Safety Board

Enclosure
L-14-289

FirstEnergy Nuclear Operating Company, Davis-Besse Unit 1,
Pressure and Temperature Limits Report for the Earlier of 32 Effective Full Power Years
or April 22, 2017, Revision 2

(Nine Pages Follow)

FIRSTENERGY NUCLEAR OPERATING COMPANY

DAVIS-BESSE UNIT 1

PRESSURE AND TEMPERATURE LIMITS REPORT

FOR THE EARLIER OF 32 EFFECTIVE FULL POWER YEARS OR APRIL 22, 2017

Revision 2

Prepared by: Thomas M Gulvas 2/25/2014
Thomas G. Gulvas

Reviewed by: J Kemp 2/28/14
Jessica Kemp

Approved by: Dennis R Blakely Date: 2/28/14
Dennis R. Blakely

FirstEnergy Nuclear Operating Company
Davis-Besse Unit 1
Pressure and Temperature Limits Report
for the Earlier of 32 Effective Full Power Years or April 22, 2017

1.0 Introduction

This Pressure and Temperature Limits Report (PTLR) provides the information required by Davis-Besse Nuclear Power Station (DBNPS) Technical Specification 5.6.4 to ensure that the Reactor Coolant System (RCS) pressure boundary is operated in accordance with its design. The limits provided are valid to 32 Effective Full Power Years (EFPY) of operation or April 22, 2017, whichever occurs first.

The PTLR provides the RCS Operating Limits in Section 2.0, which satisfies Technical Specification 5.6.4.a. The Analytical Methods used to develop the limits, including determination of the vessel neutron fluence, are provided in Section 3.0, fulfilling Technical Specification 5.6.4.b. The information and formatting of Section 3 follows the guidance of Attachment 1 to Generic Letter 96-03. The PTLR requirements are provided in Section 4.0 of the report, fulfilling Technical Specification 5.6.4.c.

Revision 0 was the initial issue of the 32 EFPY PTLR after issuance of License Amendment 282, which authorized use of new methodologies.

Revision 1 is re-issuing the 32 EFPY Pressure-Temperature limits to include the limits for the Reactor Vessel Closure Head (RVCH) installed in October 2011 Cycle 17 Mid-cycle Outage. The limits associated with the RVCH obtained from the Midland nuclear power plant have been removed. No methodology changes occurred in this revision.

Revision 2 is re-issuing the 32 EFPY Pressure-Temperature limits to incorporate Revision 4 of ANP-2718, "Appendix G Pressure-Temperature Limits for 52 EFPY, Using ASME Code Cases for Davis-Besse Nuclear Power Station" (Reference 5.7). Revision 4 of ANP-2718 combined the Heatup/Cooldown Curves into a single Figure. This results in the re-numbering of the In-Service Leak and Hydrostatic Tests Figure to Figure 2. No methodology changes occurred in this revision.

Revisions to the PTLR are to be submitted to the NRC after issuance.

2.0 RCS Pressure and Temperature Limits

a. The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines and ramp rates shown on Figures 1 and 2 (Reference 5.7) during heatup, cooldown, criticality, and in-service leak and hydrostatic (ISLH) testing with:

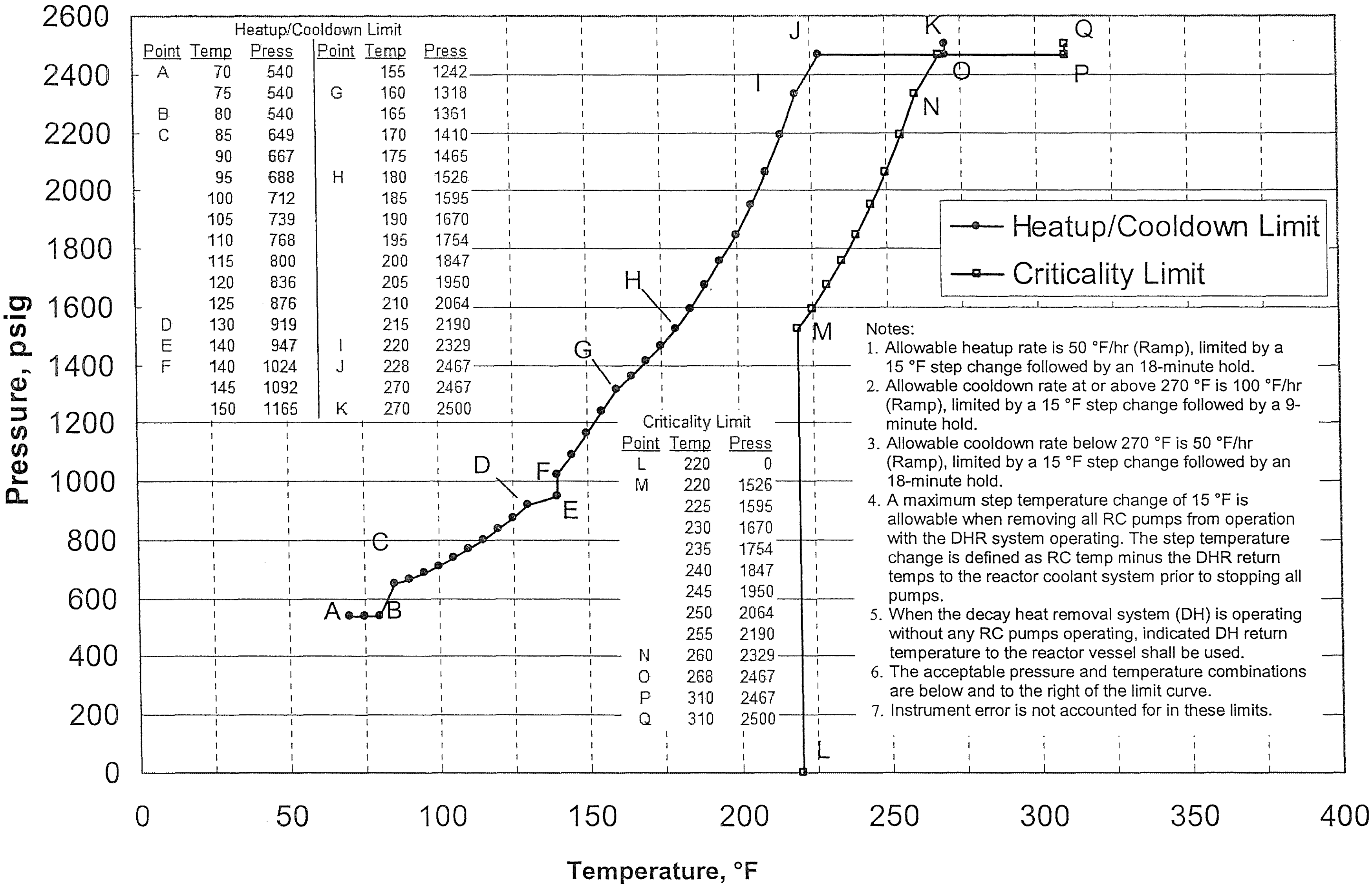
   1. A maximum heatup of 50°F in any one hour period, and

2. A maximum cooldown of 100°F in any one hour period with a cold leg temperature of $\geq$ 270°F and a maximum cooldown of 50°F in any one hour period with a cold leg temperature of < 270°F.

b. During periods of low temperature operation (T_{avg} <280 °F), Technical Specification 3.4.12 (Reference 5.3) provides additional requirements for RCS pressure and temperature limits. Those limits are maintained in the Technical Specifications because they are not determined using methods generically approved by the NRC.

Intentionally Blank

Figure 1: Composite Normal Heatup/Cooldown Limit – Both Hot Leg Pressure Taps



Heatup/Cooldown Limit

Point	Temp	Press	Point	Temp	Press
A	70	540		155	1242
	75	540	G	160	1318
B	80	540		165	1361
C	85	649		170	1410
	90	667		175	1465
	95	688	H	180	1526
	100	712		185	1595
	105	739		190	1670
	110	768		195	1754
	115	800		200	1847
	120	836		205	1950
	125	876		210	2064
D	130	919		215	2190
E	140	947	I	220	2329
F	140	1024	J	228	2467
	145	1092		270	2467
	150	1165	K	270	2500

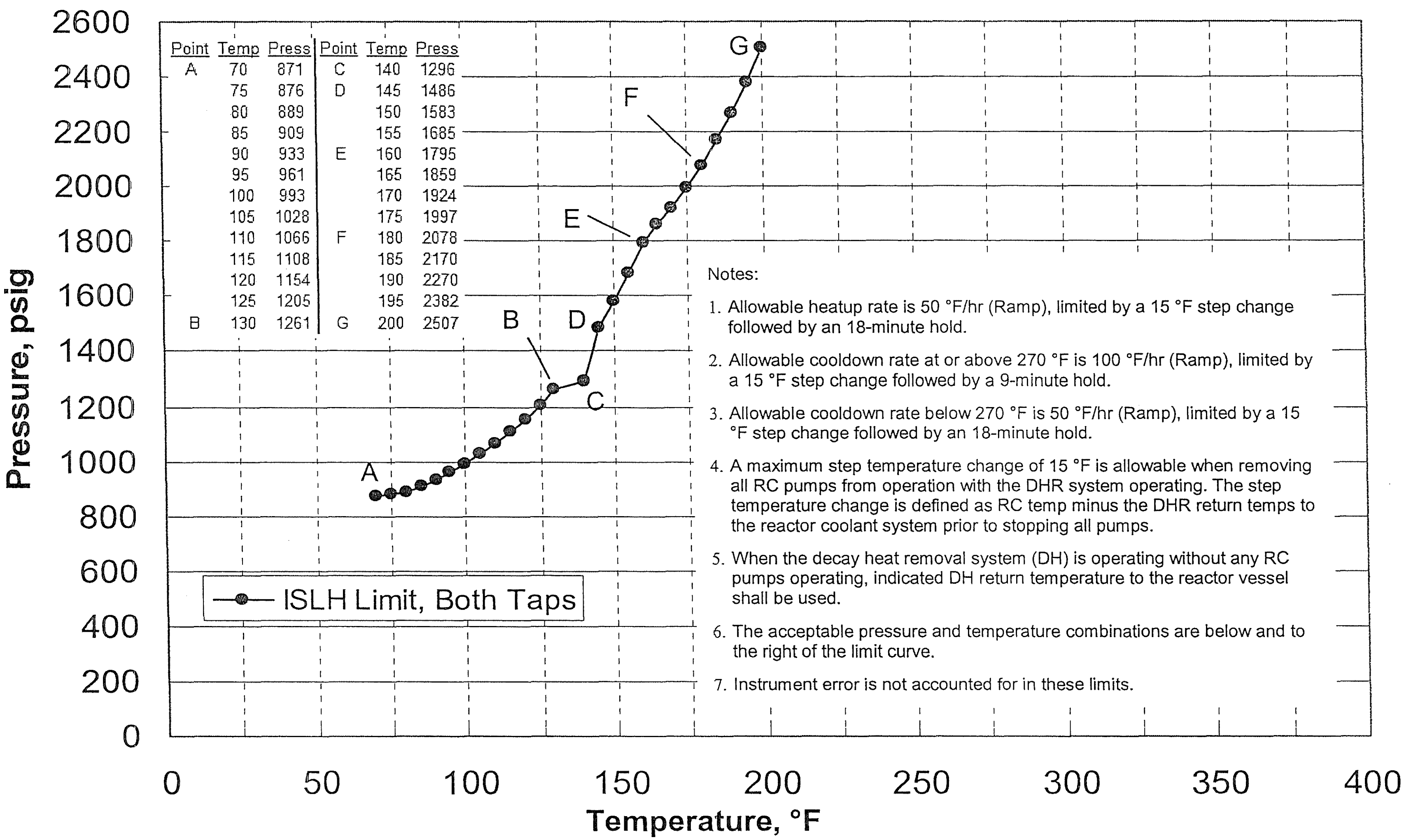
Criticality Limit

Point	Temp	Press
L	220	0
M	220	1526
	225	1595
	230	1670
	235	1754
	240	1847
	245	1950
	250	2064
	255	2190
N	260	2329
O	268	2467
P	310	2467
Q	310	2500

Notes:
1. Allowable heatup rate is 50 °F/hr (Ramp), limited by a 15 °F step change followed by an 18-minute hold.
2. Allowable cooldown rate at or above 270 °F is 100 °F/hr (Ramp), limited by a 15 °F step change followed by a 9-minute hold.
3. Allowable cooldown rate below 270 °F is 50 °F/hr (Ramp), limited by a 15 °F step change followed by an 18-minute hold.
4. A maximum step temperature change of 15 °F is allowable when removing all RC pumps from operation with the DHR system operating. The step temperature change is defined as RC temp minus the DHR return temps to the reactor coolant system prior to stopping all pumps.
5. When the decay heat removal system (DH) is operating without any RC pumps operating, indicated DH return temperature to the reactor vessel shall be used.
6. The acceptable pressure and temperature combinations are below and to the right of the limit curve.
7. Instrument error is not accounted for in these limits.

Figure 2

Reactor Coolant System Pressure-Temperature Heatup and Cooldown Limits for In-Service Leak and Hydrostatic Tests



Point	Temp	Press	Point	Temp	Press
A	70	871	C	140	1296
	75	876	D	145	1486
	80	889		150	1583
	85	909		155	1685
	90	933	E	160	1795
	95	961		165	1859
	100	993		170	1924
	105	1028		175	1997
	110	1066	F	180	2078
	115	1108		185	2170
	120	1154		190	2270
	125	1205		195	2382
B	130	1261	G	200	2507

Notes:

1. Allowable heatup rate is 50 °F/hr (Ramp), limited by a 15 °F step change followed by an 18-minute hold.
2. Allowable cooldown rate at or above 270 °F is 100 °F/hr (Ramp), limited by a 15 °F step change followed by a 9-minute hold.
3. Allowable cooldown rate below 270 °F is 50 °F/hr (Ramp), limited by a 15 °F step change followed by an 18-minute hold.
4. A maximum step temperature change of 15 °F is allowable when removing all RC pumps from operation with the DHR system operating. The step temperature change is defined as RC temp minus the DHR return temps to the reactor coolant system prior to stopping all pumps.
5. When the decay heat removal system (DH) is operating without any RC pumps operating, indicated DH return temperature to the reactor vessel shall be used.
6. The acceptable pressure and temperature combinations are below and to the right of the limit curve.
7. Instrument error is not accounted for in these limits.

3.0 Analytical Methods

3.1 The limits provided in Section 2 and Figures 1 and 2 are valid until the Reactor Vessel has accumulated 32 Effective Full Power Years (EFPY) of fast (E> 1 MeV) neutron fluence or April 22, 2017, whichever comes first.

3.2 The neutron fluence is calculated (Reference 5.12 with Reference 5.13) consistent with Regulatory Guide 1.190 using the NRC-approved methodology described in BAW-2241P-A (Reference 5.5). Table 1 provides the neutron fluence values used in the adjusted reference temperature calculations. The listed fluence values are based on 52 EFPY of operation. The limits in Section 2 are administratively limited as described in Section 3.1 based on the current Operating License of Davis-Besse Nuclear Power Station.

3.3 The Davis-Besse Reactor Vessel Material Surveillance Program complies with the requirements of Appendix H to 10 CFR 50 and is described in BAW-1543A (Reference 5.6). This information was approved by the NRC in the SER of Amendment 199 (Reference 5.1). The specimen capsule withdrawal schedule is contained within the supplements of the topical report. All plant specific specimen capsules have been withdrawn from the reactor vessel. The ART values were not calculated using surveillance data (Reference 5.14) since it was determined to be non-credible.

3.4 Low Temperature Overpressure Protection (LTOP) limits are addressed in Section 2.b, above, and Technical Specification 3.4.12 (Reference 5.3). Reference 5.7 discusses the methods used to determine the temperature at which LTOP must be active. The pressure limit was determined using ASME Section XI, Appendix G, as modified by the alternative rules provided in ASME Code Case N-588 and ASME Code Case N-640 (Reference 5.9).

3.5 Table 1 provides the Adjusted Reference Temperature (ART) for each reactor vessel beltline material. The ART values were calculated in accordance with Regulatory Guide 1.99, Revision 2. For welds in the reactor beltline region, the initial RT_{NDT} values used (in part) to determine ART were calculated using an alternate methodology described in the NRC-approved BAW-2308, Revisions 1-A and 2-A (Reference 5.10). 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 Reference 5.17. The NRC confirmed the limits and conditions for using the methodology were satisfied in the SER of Amendment 282 (Reference 5.8).

3.6 The Pressure-Temperature (P/T) limits of Section 2 and Figures 1, 2, and 3 (with applicability as stated in 3.1) were generated consistent with the requirements of 10 CFR 50 Appendix G and Regulatory Guide 1.99, Revision 2, using the methods described in BAW-10046A (Reference 5.4) and ASME Section XI,

Appendix G (Reference 5.9), as modified by the alternative rules provided in ASME Code Case N-588 and ASME Code Case N-640.

3.6.1 The NRC has reviewed the methods described in BAW-10046A (Reference 5.4) and approved the topical report by issuance of a Safety Evaluation Report (SER) dated April 30, 1986. Section 1.2 of BAW-10046A states that it is applicable to all current B&W nuclear steam systems.

3.6.2 ASME Code Cases N-640 and N-588 have been incorporated into ASME Section XI, Appendix G, 2003 Addenda, which are the edition and addenda codified in 10 CFR 50.55a (effective May 27, 2008) and thus may be used per NRC Regulatory Issue Summary (RIS) 2004-04. Specific approval for application at DBNPS is included in Ref. 5.8.

3.7 The minimum temperature requirements of 10 CFR 50, Appendix G are included on Figures 1 and 2. Figure 3 provides the In-Service Leak and Hydrostatic (ISLH) Test Limits. These limits were calculated in accordance with the requirements of 10 CFR 50, Appendix G and ASME Code Section XI, Appendix G, 1995 Edition, with Addenda through 1996 and ASME Code Cases N-588 and N-640.

3.8 Davis-Besse has removed more than two surveillance capsules. The capsule test results have been evaluated and found to be non-credible (Reference 5.14). Consequently, ART calculations are not based on the surveillance data. The Measured ΔRT_{NDT} – Predicted ΔRT_{NDT} data scatter was less than 2σ, so the Regulatory Guide 1.99, Rev. 2 Chemistry Table values used in the ART calculations are conservative.

4.0 PTLR Requirements

4.1 The PTLR has been prepared in accordance with the requirements of Technical Specification 5.6.4 (see Reference 5.11). The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. Davis-Besse will continue to meet the requirements of 10 CFR 50, Appendix G, and any changes to the Davis-Besse P/T limits will be generated in accordance with the NRC approved methodologies described in TS 5.6.4.

Table1: Davis-Besse Nuclear Power Station Reactor Vessel Beltline Region Data (Applicable as noted in Section 3.1)

Reactor Vessel Location	Material Identification	Fluence @ 52 EFPY (Wetted Surface) (n/cm^2) (E> 1 MeV)	ART @ ¼ T (°F) @52 EFPY (Note 1)	ART @ ¾ T (°F) @52 EFPY (Note 1)	Limiting Mat'l? (Yes/No)	RT_{PTS} (°F) (Note 2)
Nozzle Belt Forging	ADB 203	2.29E+18	74.8	64.8	No	81.2
Nozzle Belt to Upper Shell Weld (ID 9%)	WF-232	2.29E+18	Note 3	Note 3	No	157.9
Nozzle Belt to Upper Shell Weld (OD 91%)	WF-233	2.29E+18	100.4*	67.8*	No	Note 4
Upper Shell Forging	AKJ 233	1.69E+19	71.8	57.3	No	79.4
Upper Shell to Lower Shell Weld	WF-182-1	1.69E+19	156.2*	106.4*	Yes	182.2*
Lower Shell Forging	BCC 241	1.70E+19	89.9	78.8	Yes	95.7

Note 1: Reported ART values are based on Regulatory Guide 1.99, Revision 2 (Ref. 5.15). P/T Limit calculation was based on a temperature value which is more conservative than the listed ART value. (Ref. 5.13)
Note 2: Values from Ref. 5.16, which are based on the location specific clad to vessel interface fluence at 52 EFPY.
Note 3: This weld material does not extend out to the ¼T or ¾T location.
Note 4: This weld material is not present at the clad to vessel interface, so RT_{PTS} does not apply to it.
* Based on the initial RT_{NDT} provided in the NRC Safety Evaluation Reports to BAW-2308, Rev. 1-A and 2-A (Ref. 5.10).

5.0 References

5.1 Safety Evaluation by the NRC Office of Nuclear Reactor Regulation Related to Amendment No. 199 to Facility Operating License No. NPF-3 Davis-Besse Nuclear Power Station, Unit No. 1, attached to correspondence dated July 20, 1995.

5.2 Technical Specification 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)."

5.3 Technical Specification 3.4.12, "Low Temperature Overpressure Protection."

5.4 BAW-10046A, Revision 2 "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50 Appendix G."

5.5 BAW-2241P-A, "Fluence and Uncertainty Methodologies," dated April 1999.

5.6 BAW-1543A, "Master Integrated Reactor Vessel Material Surveillance Program."

5.7 ANP-2718, Revision 4, "Appendix G Pressure-Temperature Limits for 52 EFPY, Using ASME Code Cases for Davis-Besse Nuclear Power Station," dated December 2013.

5.8 Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 282 to Facility Operating License No. NPF-3, FirstEnergy Nuclear Operating Company Davis-Besse Nuclear Power Station, Unit No. 1, (FENOC Ltr. R11-030), dated 01/28/2011.

5.9 ASME Code Section XI, Appendix G, as modified by the alternate rules provided in ASME Code Case N-640 and ASME Code Case N-588. ASME Code Cases N-640 and N-588 have subsequently been incorporated into ASME Section XI, Appendix G, 2003 Addenda, which are the edition and addenda codified in 10 CFR 50.55a (effective May 27, 2008).

5.10 BAW-2308, Revision 1-A and Revision 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials," dated August 2005 (1-A) and March 2008 (2-A).

5.11 Calculation C-NSA-064.02-037, Revision 1, "Davis-Besse 52 EFPY PT Limits – Chalon RV Closure Head," dated 9/23/2011.

5.12 AREVA Report 86-9015129-000, "DB1 – Cycles 13-15 Fluence Analysis Report," dated 4/21/2006.

5.13 AREVA Report 51-9123331-000, "Davis-Besse – EOL Fluence Reconciliation," dated 10/8/2009.

5.14 AREVA Document 32-9031157-000, "Davis-Besse Revised ART Values at 52 EFPY," dated 9/20/2006.

5.15 AREVA Document 32-9017744-003, "Davis-Besse ART Values at 52 EFPY," dated 10/29/2009.

5.16 AREVA Document 32-9123247-000, "RT_{PTS} Values of Davis-Besse Unit 1 for 52 EFPY, Including Extended Beltline," dated 11/12/09.

5.17 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," (FENOC Ltr. R10-298) dated December 14, 2010.