

March 7, 2011
L-11-085

10 CFR 50.90

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

SUBJECT:
Davis-Besse Nuclear Power Station
Docket No. 50-346, License No. NPF-3
Submittal of the Unapproved Pressure and Temperature Limits Report
(TAC Nos. ME1127 and ME1128)

By letter dated April 15, 2009 (Accession No. ML091130228), and supplemented by letters dated December 18, 2009 (Accession No. ML093570103), August 26, 2010 (Accession No. ML102440105) and October 8, 2010 (Accession No. ML102861221), the FirstEnergy Nuclear Operating Company (FENOC) submitted to the Nuclear Regulatory Commission (NRC) a license amendment request and an exemption request for the Davis-Besse Nuclear Power Station (DBNPS). The proposed amendment would incorporate alternate methodologies for developing the DBNPS reactor pressure vessel pressure and temperature limit curves. The proposed exemption would exempt DBNPS from certain requirements contained in 10 CFR 50.61 and 10 CFR Part 50, Appendix G.

By electronic mail dated January 10, 2011 (Accession No. ML110100757), FENOC submitted, as requested by the NRC staff, an unapproved version of the DBNPS pressure and temperature limits report (PTLR) based upon the use of the methodologies requested in the proposed license amendment request and the exemption request.

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NRK

This submittal provides the unapproved PTLR for inclusion onto the DBNPS docket. The unapproved version is being provided for administrative purposes only (for example, report format and type of information to be provided). The technical data included in the report is preliminary data and is subject to change. Submittal of the approved version, which may differ from the unapproved version, will be performed in accordance with Technical Specification 5.6.4, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)."

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) 761-6071.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 7, 2011.

Sincerely,



Kendall W. Byrd
Director, Site Performance Improvement

Attachment:
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

cc: NRC Region III Administrator
NRC Project Manager
NRC Resident Inspector
Executive Director, Ohio Emergency Management Agency,
State of Ohio (NRC Liaison)
Utility Radiological Safety Board

**Attachment
L-11-085**

**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
(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

Prepared by: Dennis Blakely NOT APPROVED

Reviewed by: Kevin Burnworth NOT APPROVED

Approved by: Kevin Zellers NOT APPROVED

FirstEnergy Nuclear Operating Company
Davis-Besse Unit 1
32 Effective Full Power Years
Pressure and Temperature Limit Report

1.0 Introduction

This Pressure and Temperature Limit 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 Pressure Boundary is operated in accordance with its design.

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.

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, 2, and 3 (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 $\geq 270^\circ\text{F}$ and a maximum cooldown of 50°F in any one hour period with a cold leg temperature of $< 270^\circ\text{F}$.
- b. During periods of low temperature operation ($T_{\text{avg}} < 280^\circ\text{F}$), 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.

Figure 1: Composite Normal Heatup/Cooldown Limit – Hot Leg “A” Pressure Tap

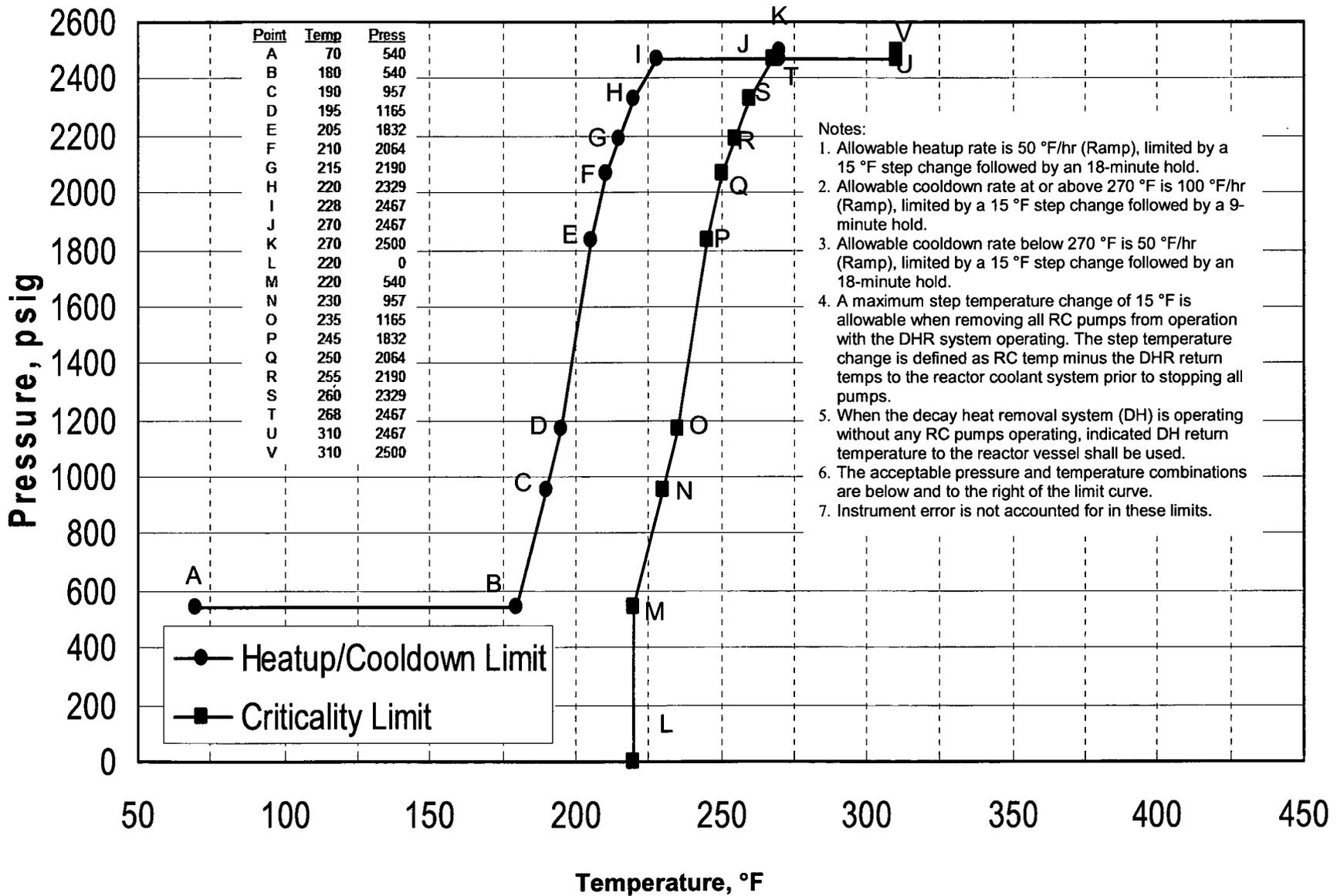


Figure 2: Composite Normal Heatup/Cooldown Limit – Hot Leg “B” Pressure Tap

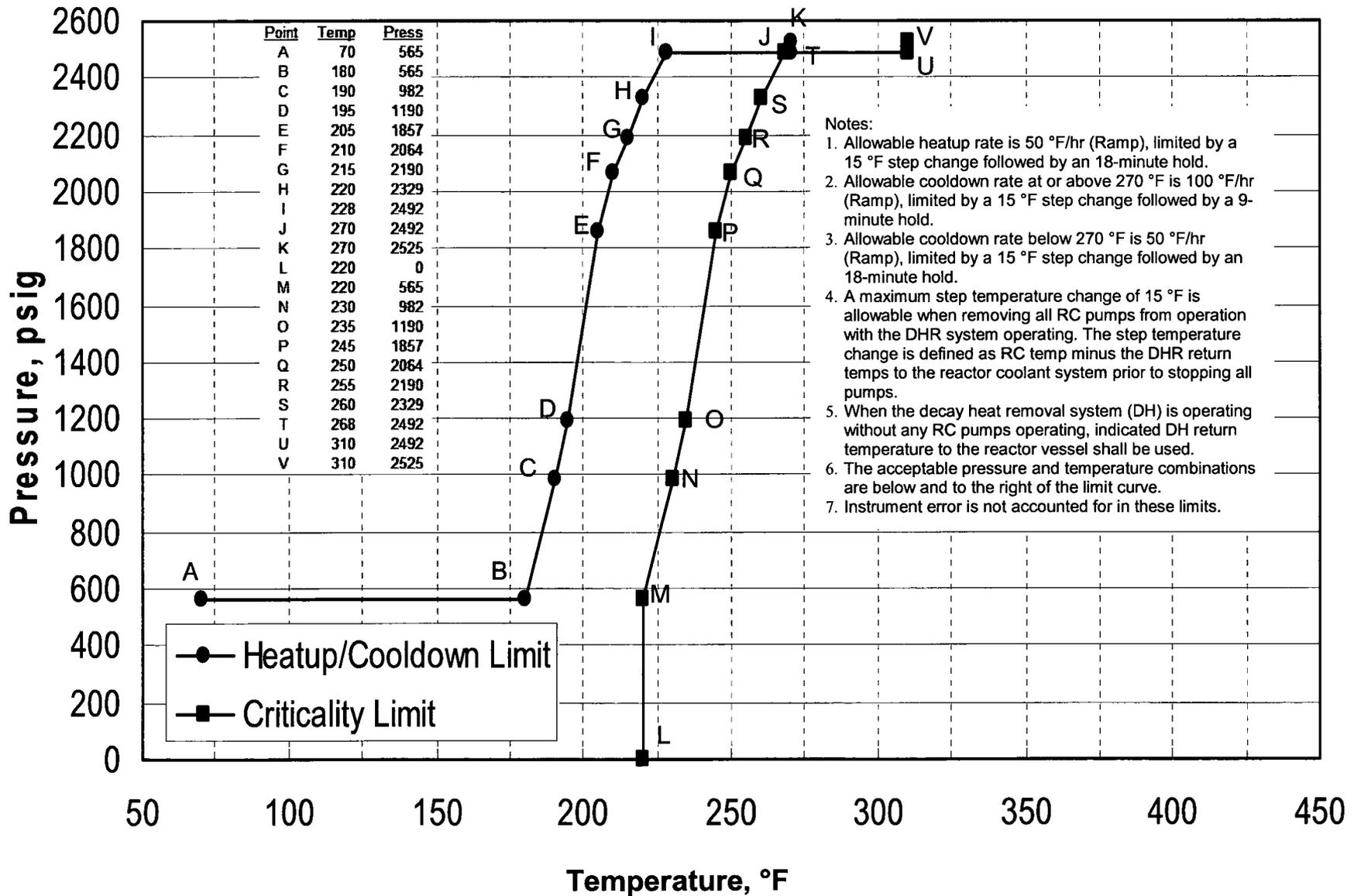
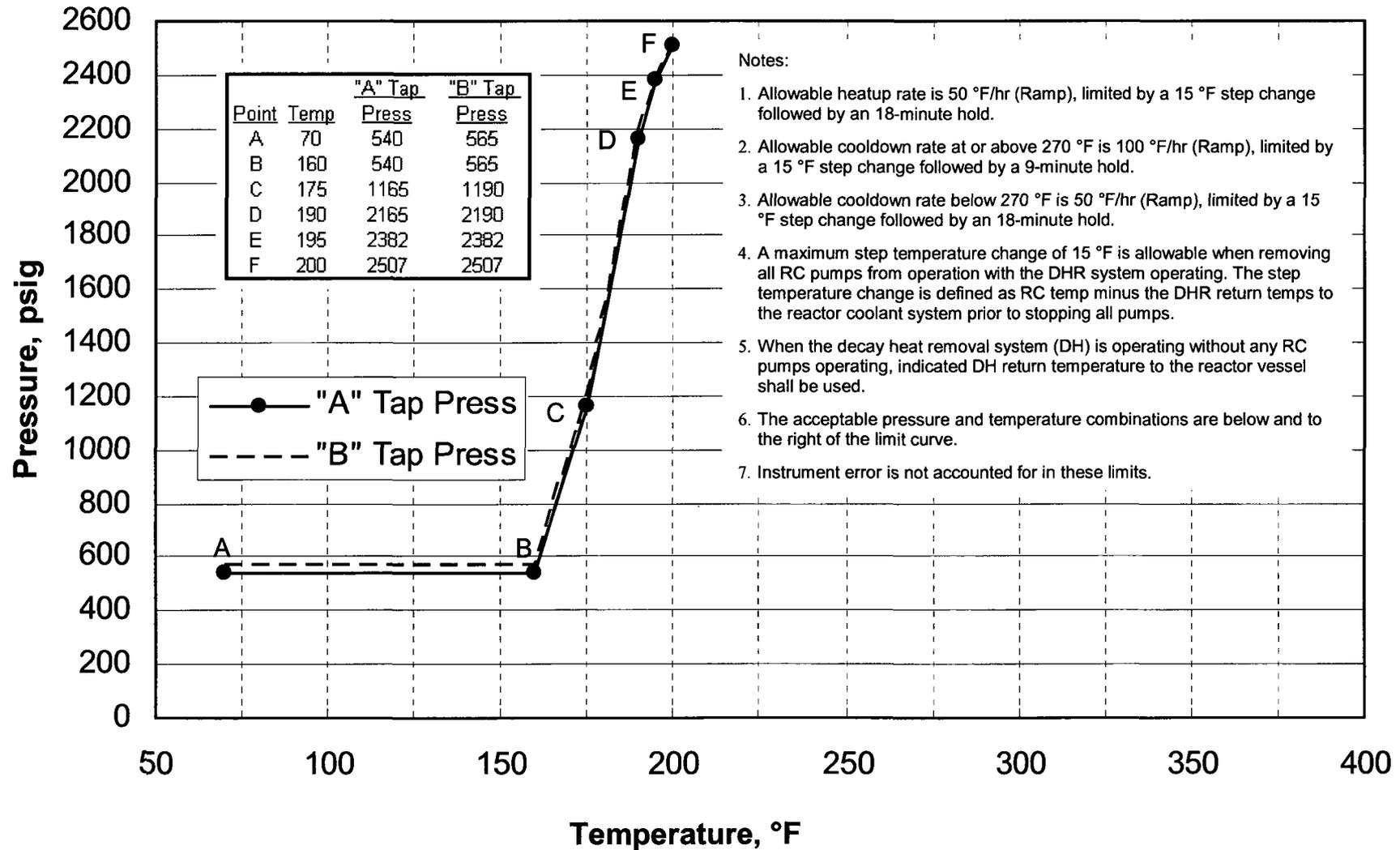


Figure 3
Reactor Coolant System Pressure-Temperature Heatup and
Cooldown Limits for In-Service Leak and Hydrostatic Tests



3.0 Analytical Methods

- 3.1 The limits provided in Section 2 and Figures 1, 2, and 3 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 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, which meets the requirements of ASME Section XI, Appendix G, 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 (Reference 5.9), as modified by the alternative rules provided in ASME Code Case N-588 and ASME Code Case N-640.
- 3.5 Table 1 provides the Adjusted Reference Temperature (ART) for each reactor 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 was calculated using an alternate methodology described in the NRC-approved BAW-2308, Revisions 1-A and 2-A (Reference 5.10). As stated in the NRC Safety Evaluations for the BAW-2308 topical reports, an exemption request to use the alternate initial RT_{NDT} values must be submitted to the NRC. The exemption was granted, and the limits and conditions for using the methodology were confirmed by the NRC to be satisfied in the SER of Amendment ??? (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 10CFR50, 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 10CFR50, 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 Adjusted ART – Predicted ART 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 (Note1)	ART @ ¾ T (°F) @52 EFPY (Note1)	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	118.3
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 BAW-2308, Rev. 1A and 2A with NRC Safety Evaluation Reports (Ref. 5.8)

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 3, "Appendix G Pressure-Temperature Limits for 52 EFPY, Using ASME Code Cases for Davis-Besse Nuclear Power Station," dated August 2010.
- 5.8 Safety Evaluation by the NRC Office of Nuclear Reactor Regulation Related to License Amendment Request 08-034 to Facility Operating License No. NPF-3 Davis-Besse Nuclear Power Station, Unit No. 1.
- 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 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 0, "Davis-Besse 52 EFPY PT Limits – Midland RV Closure Head," dated ????
- 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.