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January 10, 2023
L-22-253

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
Submittal of Pressure and Temperature Limits Report, Revision 5

In accordance with Technical Specification 5.6.4, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," Energy Harbor Nuclear Corp. hereby submits Revision 5 of the PTLR for the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS). Revision 5, which was approved on November 30, 2022, and made effective on December 20, 2022, corrects errors identified in Revision 4 of the report. Minor editorial changes were made in Figure 1 and Section 2.0. Additionally, the table in Figure 2 incorrectly listed pressure data in the temperature data column; however, the associated graph was correct. This item is identified and tracked in the Corrective Action Program.

There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Phil H. Lashley, Manager - Fleet Licensing, at (330) 696-7208.

Sincerely,

Terry J. Brown

Enclosure:
Pressure and Temperature Limits Report for Up to 43.5 Effective Full Power Years, Revision 5

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

Enclosure

Pressure and Temperature Limits Report for Up to 43.5 Effective Full Power Years, Revision 5

L-22-253

(12 pages follow)

ENERGY HARBOR NUCLEAR OPERATING COMPANY

DAVIS-BESSE UNIT 1

PRESSURE AND TEMPERATURE LIMITS REPORT

FOR UP TO 43.5 EFFECTIVE FULL POWER YEARS

Revision 5

Prepared by: _JR M_ Date: 11/30/22
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Reviewed by: _Michael L Nelson_ Date: 11/30/22
Michael L. Nelson

Brian A. Kanney
Approved by: _BAK_ Date: 30NOV2022
Brian A. Kanney

EnergyHarbor Nuclear Operating Company
Davis-Besse Unit 1
Pressure and Temperature Limits Report
for up to 43.5 Effective Full Power Years

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 43.5 Effective Full Power Years (EFPY).

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.

Revision 3 removes the restriction of exceeding the operating limit date of April 22, 2017 which was included in earlier revisions. This change is the result of the Nuclear Regulatory Commission issuing a renewed operating license to Davis-Besse which extends the period of operation to midnight April 22, 2037, which made 32 EFPY limiting. This change also corrects an administrative error that existed in the previous revision where Figure 2 was combined with Figure 1 and the original Figure 3 was designated as Figure 2, however Figure 3 was still referred to in the body of the document.

Revision 4 is the initial issue of the 43.5 EFPY PTLR. CR-2019-03982, Reactor Vessel 52 EFPY Projected Neutron Fluence (Exposure) Higher Than Previous Projections (Reference 5.17), identified that due to the reactor vessel (RV) fast neutron fluence accumulating faster than previously predicted, the RV would reach the analyzed fluence

sometime prior to 52 Effective Full Power Years (EFPY) of operation. Document 32-9300671-000, Davis-Besse Reactor Vessel Embrittlement Fluence Reconciliation Through 60 Years (Reference 5.16) and CR 2020-07840 "Revised Vessel P-T Curves Limited to 43 EFPY" (Reference 5.18), has determined that the most limiting part of the RV will reach the analyzed fluence at approximately 43.5 EFPY. Data is based on fluence projections of Fuel Cycles 18 and 19. No methodology changes occurred in this revision.

Revision 5 is re-issuing the 43.5 EFPY Pressure and Temperature Limits Report to remove two data points in Figure 1 (Heatup Limit) at 226 °F and to incorporate the correct temperature data table in Figure 2. Consistent use of units was incorporated into Section 2.0 "RCS Pressure and Temperature Limits". 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

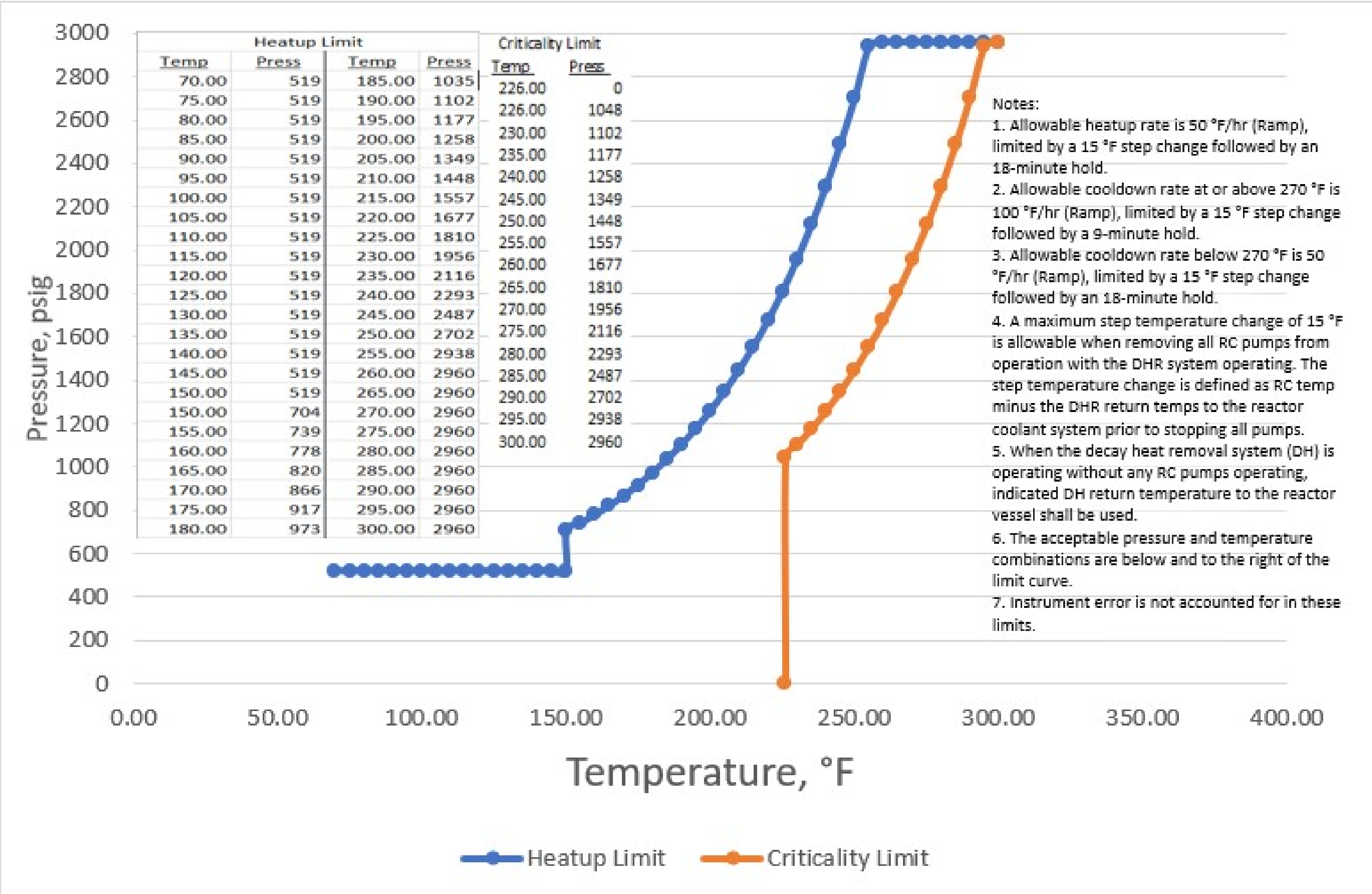
a. The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines and ramp rates shown on Figures 1, 2, 3 and 4 (Reference 5.7) during heatup, cooldown, criticality, and in-service leak and hydrostatic (ISLH) testing with:

   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.

Figure 1: Reactor Coolant System Pressure-Temperature Heatup and Criticality Limits

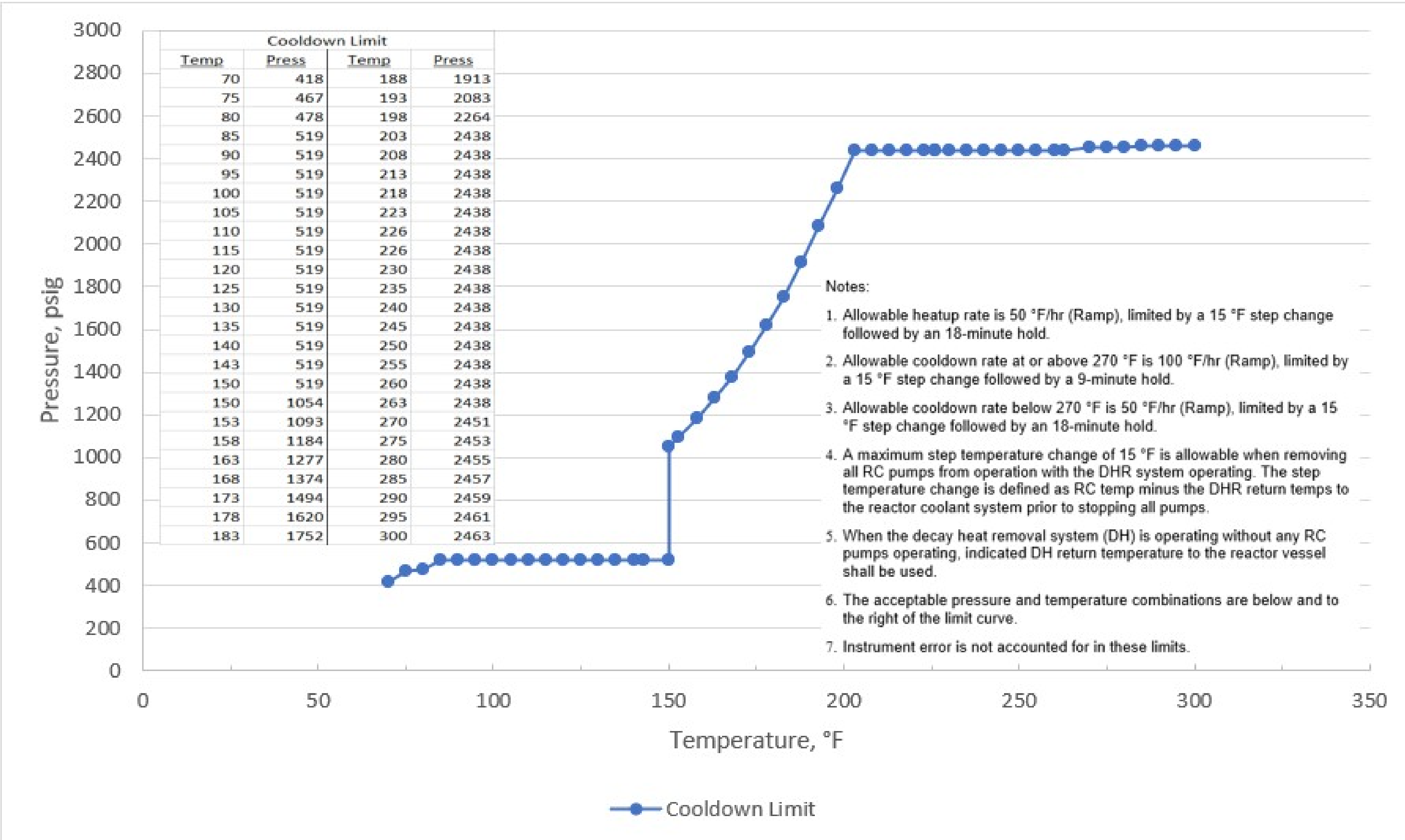


Heatup Limit			
Temp	Press	Temp	Press
70.00	519	185.00	1035
75.00	519	190.00	1102
80.00	519	195.00	1177
85.00	519	200.00	1258
90.00	519	205.00	1349
95.00	519	210.00	1448
100.00	519	215.00	1557
105.00	519	220.00	1677
110.00	519	225.00	1810
115.00	519	230.00	1956
120.00	519	235.00	2116
125.00	519	240.00	2293
130.00	519	245.00	2487
135.00	519	250.00	2702
140.00	519	255.00	2938
145.00	519	260.00	2960
150.00	519	265.00	2960
150.00	704	270.00	2960
155.00	739	275.00	2960
160.00	778	280.00	2960
165.00	820	285.00	2960
170.00	866	290.00	2960
175.00	917	295.00	2960
180.00	973	300.00	2960

Criticality Limit	
Temp	Press
226.00	0
226.00	1048
230.00	1102
235.00	1177
240.00	1258
245.00	1349
250.00	1448
255.00	1557
260.00	1677
265.00	1810
270.00	1956
275.00	2116
280.00	2293
285.00	2487
290.00	2702
295.00	2938
300.00	2960

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.

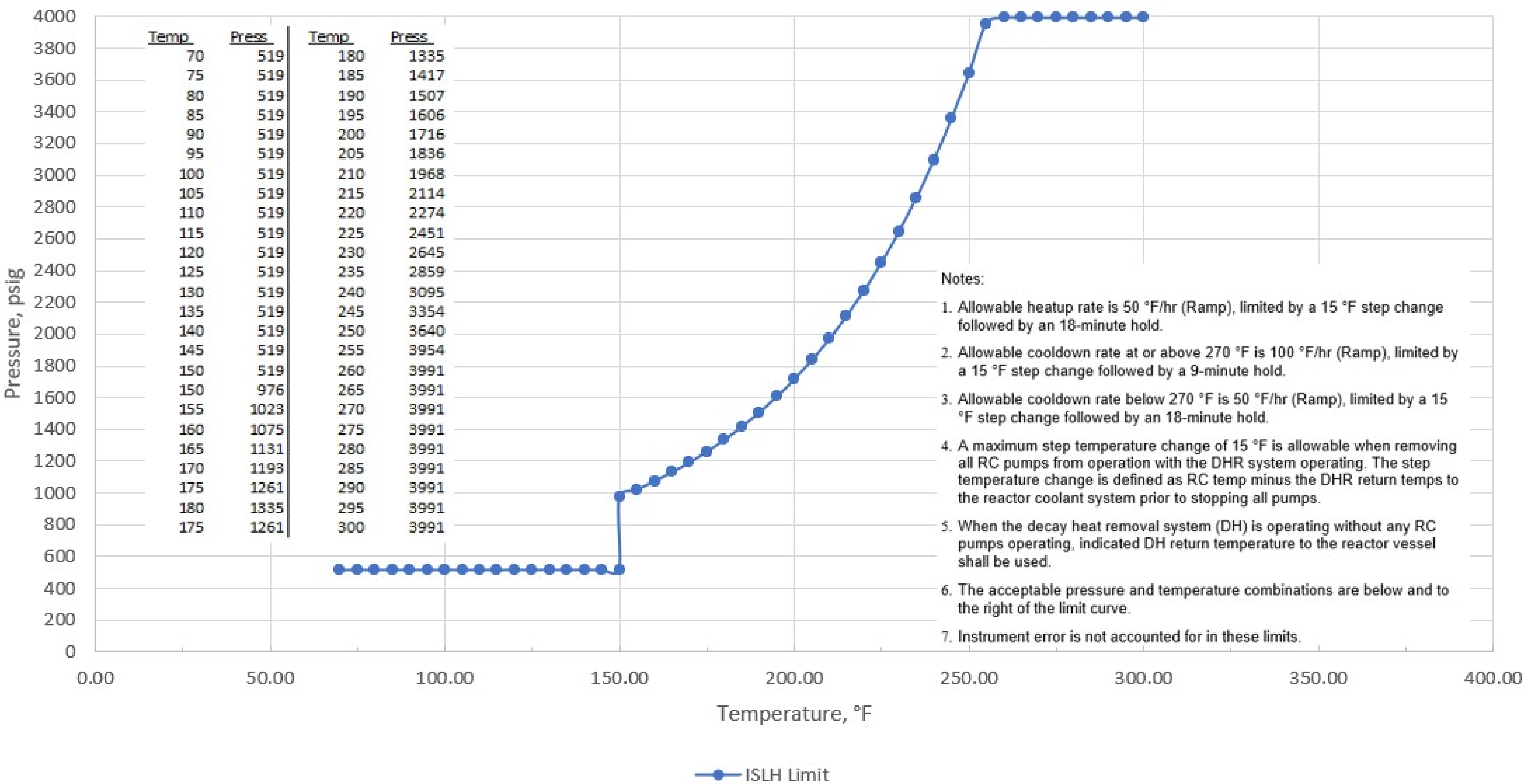
Figure 2: Reactor Coolant System Pressure-Temperature Cooldown Limits



Cooldown Limit

Temp	Press	Temp	Press
70	418	188	1913
75	467	193	2083
80	478	198	2264
85	519	203	2438
90	519	208	2438
95	519	213	2438
100	519	218	2438
105	519	223	2438
110	519	226	2438
115	519	226	2438
120	519	230	2438
125	519	235	2438
130	519	240	2438
135	519	245	2438
140	519	250	2438
143	519	255	2438
150	519	260	2438
150	1054	263	2438
153	1093	270	2451
158	1184	275	2453
163	1277	280	2455
168	1374	285	2457
173	1494	290	2459
178	1620	295	2461
183	1752	300	2463

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

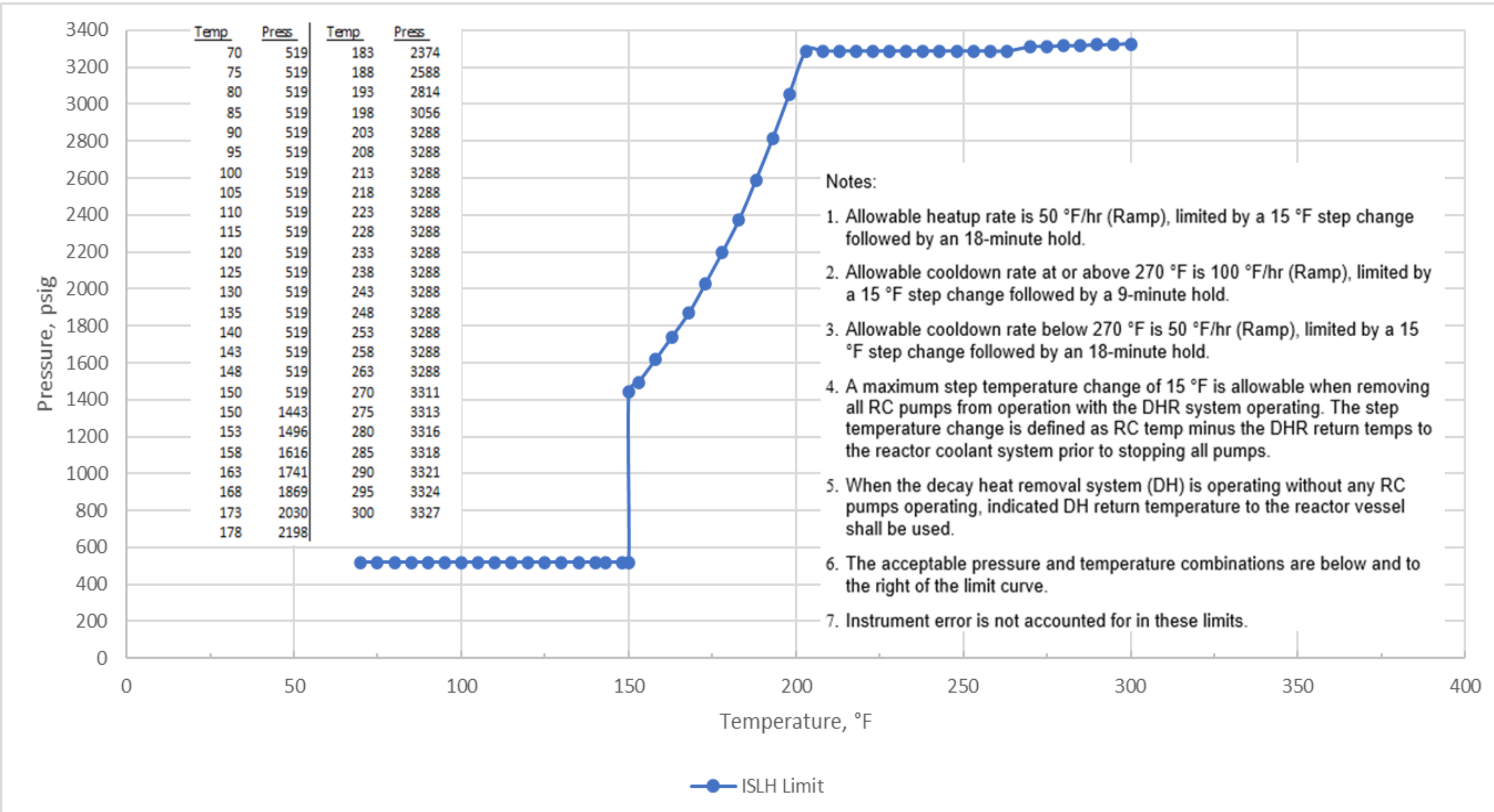


Temp	Press	Temp	Press
70	519	180	1335
75	519	185	1417
80	519	190	1507
85	519	195	1606
90	519	200	1716
95	519	205	1836
100	519	210	1968
105	519	215	2114
110	519	220	2274
115	519	225	2451
120	519	230	2645
125	519	235	2859
130	519	240	3095
135	519	245	3354
140	519	250	3640
145	519	255	3954
150	519	260	3991
150	976	265	3991
155	1023	270	3991
160	1075	275	3991
165	1131	280	3991
170	1193	285	3991
175	1261	290	3991
180	1335	295	3991
175	1261	300	3991

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



Temp	Press	Temp	Press
70	519	183	2374
75	519	188	2588
80	519	193	2814
85	519	198	3056
90	519	203	3288
95	519	208	3288
100	519	213	3288
105	519	218	3288
110	519	223	3288
115	519	228	3288
120	519	233	3288
125	519	238	3288
130	519	243	3288
135	519	248	3288
140	519	253	3288
143	519	258	3288
148	519	263	3288
150	519	270	3311
150	1443	275	3313
153	1496	280	3316
158	1616	285	3318
163	1741	290	3321
168	1869	295	3324
173	2030	300	3327
178	2198		

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, 2, 3, and 4 are valid until the Reactor Vessel has accumulated 43.5 Effective Full Power Years (EFPY) of fast (E> 1 MeV) neutron fluence (Reference 5.16).

3.2 The neutron fluence is calculated (Reference 5.7) 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.

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 5.15. 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, 3, and 4 (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, 2, 3 and 4. Figures 3 and 4 provide 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 for plant implementation). 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	Wetted Inner Surface Fluence, n/cm² (E > 1.0 MeV) Current Licensing Basis (CLB) @52 EFPY (Ref. 5.16)	Projected EFPY to Reach CLB Fluence (Ref. 5.16)	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	2.20E+18	47.4	67.1	58.7	No	82.0
Nozzle Belt to Upper Shell Weld (ID 9%)	WF-232	2.17E+18	45.9	Note 3	Note 3	No	120.5
Nozzle Belt to Upper Shell Weld (OD 91%)	WF-233	2.17E+18	45.9	98.3	66.2	No	Note 4
Upper Shell Forging	AKJ 233	1.64E+19	51.4	71.4	56.9	No	79.2
Upper Shell to Lower Shell Weld	WF-182-1	1.64E+19	51.4	150.9	101.2	Yes	181.1
Lower Shell Forging	BCC 241	1.64E+19	51.4	89.5	78.4	No	95.5

Note 1: Reported ART values (Ref. 5.7) are based on Regulatory Guide 1.99, Revision 2.
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.
Note 5: BWW 279/BWW 249 (Inlet Nozzle Forgings (at Lower Nozzle Belt to Inlet Nozzle Forging weld)) and WF-232/WF-233 (Lower Nozzle Belt to Inlet Nozzle Forging Welds) are projected to reach the CLB fluence at 43.5 EFPY (Ref. 5.16).

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, Revision 339, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)."

5.3 Technical Specification 3.4.12, Revision 339, "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 7, "Appendix G Pressure-Temperature Limits for 52 EFPY for Davis-Besse Nuclear Power Station FirstEnergy Nuclear Operating Company," dated April 2019.

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 3, "Davis-Besse 43.5 EFPY Pressure-Temperature Limits," dated 7/7/2021.

5.12 Not used

5.13 Not used

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

5.15 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.

5.16 AREVA Document 32-9300671-000, "Davis-Besse Reactor Vessel Embrittlement Fluence Reconciliation Through 60 Years," dated 12/10/2019.

5.17 CR 2019-03982 "Reactor Vessel 52 EFPY Projected Neutron Fluence (Exposure) Higher than Previous Projection," dated 4/30/2019.

5.18 CR 2020-07840 "Revised Vessel P-T Curves Limited to 43 EFPY," dated 10/9/2020