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December 9, 2013
L-13-360

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

SUBJECT:
Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
Pressure and Temperature Limits Report Revision

Pursuant to the requirements of Beaver Valley Power Station, Unit No. 2 (BVPS-2) Technical Specification (TS) 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," FirstEnergy Nuclear Operating Company (FENOC) hereby submits the BVPS-2 PTLR, Revision 6. TS Section 5.6.4.c requires that the PTLR be provided to the Nuclear Regulatory Commission (NRC) upon issuance for any revision or supplement thereto.

The BVPS-2 pressure-temperature (P-T) limit curves for normal heatup and cooldown of the primary reactor coolant system were previously developed for 22 effective full power years (EFPY), 30 EFPY, 40 EFPY, and 54 EFPY. The enclosed BVPS-2 PTLR, Revision 6, revised the PTLR P-T limit curves from 22 EFPY to 30 EFPY.

There are no regulatory commitments contained in this letter. 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,

E. A. L.

Eric A. Larson

Enclosure:
Beaver Valley Power Station, Unit No. 2, Pressure and Temperature Limits Report, Revision 6

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site Representative (BRP/DEP)

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Beaver Valley Power Station, Unit No. 2
Pressure and Temperature Limits Report, Revision 6
(29 Pages Follow)

5.0 ADMINISTRATIVE CONTROLS

5.2 Pressure and Temperature Limits Report

BVPS-2 Technical Specification to PTLR Cross-Reference			
Technical Specification	PTLR		
	Section	Figure	Table
3.4.3	5.2.1.1	5.2-1 5.2-2 5.2-3 5.2-4 5.2-5 5.2-6	N/A
3.4.6	N/A	N/A	5.2-3
3.4.7	N/A	N/A	5.2-3
3.4.10	N/A	N/A	5.2-3
3.4.12	5.2.1.2 5.2.1.3	5.2-8	5.2-3
3.5.2	N/A	N/A	5.2-3

BVPS-2 Licensing Requirement to PTLR Cross-Reference			
Licensing Requirement	PTLR		
	Section	Figure	Table
LR 3.1.2	N/A	N/A	5.2-3
LR 3.1.4	N/A	N/A	5.2-3
LR 3.4.6	N/A	N/A	5.2-3

5.2 Pressure and Temperature Limits Report

5.2 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

The PTLR for Unit 2 has been prepared in accordance with the requirements of Technical Specification 5.6.4. Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications (TS) and Licensing Requirements (LR) addressed, or made reference to, in this report are listed below:

1. LCO 3.4.3 Reactor Coolant System Pressure and Temperature (P/T) Limits,
2. LCO 3.4.6 RCS Loops - MODE 4,
3. LCO 3.4.7 RCS Loops - MODE 5, Loops Filled,
4. LCO 3.4.10 Pressurizer Safety Valves,
5. LCO 3.4.12 Overpressure Protection System (OPPS),
6. LCO 3.5.2 ECCS - Operating,
7. LR 3.1.2 Boration Flow Paths - Operating,
8. LR 3.1.4 Charging Pump - Operating, and
9. LR 3.4.6 Pressurizer Safety Valve Lift Involving Loop Seal or Water Discharge

5.2.1 Operating Limits

The PTLR limits for Beaver Valley Power Station (BVPS) Unit 2 were developed using a methodology specified in the Technical Specifications. The methodology listed in Reference 1 was used with two exceptions:

a) Use of ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limits for Section XI, Division 1," and

b) Use of methodology of the 1996 version of ASME Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

5.2.1.1 RCS Pressure and Temperature (P/T) Limits (LCO 3.4.3)

The RCS temperature rate-of-change limits defined in Reference 14 are:

a. A maximum heatup of 60°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period, and

5.2 Pressure and Temperature Limits Report

c. A maximum temperature change of less than or equal to 5°F in any one hour period during inservice hydrostatic testing operations above system design pressure.

The RCS P/T limits for heatup, leak testing, and criticality are specified by Figure 5.2-1 and Table 5.2-1. The RCS P/T limits for cooldown are shown in Figures 5.2-2 through 5.2-6 and Table 5.2-2. These limits are defined in Reference 14. Consistent with the methodology described in Reference 1, including the exceptions as noted in Section 5.2.1, the RCS P/T limits for heatup and cooldown shown in Figures 5.2-1 through 5.2-6 are provided without margins for instrument error. The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in 10 CFR 50, Appendix G. The heatup and cooldown curves also include the effect of the reactor vessel flange.

The P/T limits for core operation (except for low power physics testing) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding P/T curve for heatup and cooldown.

The pressure-temperature limit curve shown in Figure 5.2-7 was developed for the limiting ferritic steel component within an isolated reactor coolant loop. The limiting component is the steam generator channel head to tubesheet region. This figure provides the ASME III, Appendix G limiting curve which is used to define operational bounds, such that when operating with an isolated loop the analyzed pressure-temperature limits are known. The temperature range provided bounds the expected operating range for an isolated loop and Code Case N-640.

Reference 13 provides an updated surveillance capsule credibility evaluation, updated Position 2.1 chemistry factor values, and an updated fluence evaluation. Therefore, the applicability of the P/T limit curves (Reference 14) was assessed based on the revised information. Taking into account the updated surveillance data credibility evaluation, the Position 2.1 chemistry factor values, and the fluence analysis summarized in Reference 13, the limiting material for the current BVPS-2 P/T limits continues to be the intermediate shell plate B9004-1 at 30 EFPY.

5.2 Pressure and Temperature Limits Report

Since the adjusted reference temperature (ART) calculation is not based on surveillance data for this limiting material, only a fluence comparison is needed in order to assess the applicability of the existing curves. Using the fluence analysis provided in Table 5-1 of Reference 13, the maximum neutron fluence value at 30 EFPY is 3.03×10^{19} n/cm^2 (E > 1.0 MeV). This value was calculated by interpolating the fluence at the 0° azimuthal position for BVPS-2 from the end of Cycle 15 to the fluence value at the future projection out to 32 EFPY. The fluence of 3.39×10^{19} n/cm^2 (E > 1.0 MeV) used to develop the 30 EFPY P/T limit curves generated as a result of the Capsule X analysis (Reference 12), is more conservative than the updated fluence of 3.03×10^{19} n/cm^2 (E > 1.0 MeV).

5.2.1.2 Overpressure Protection System (OPPS) Setpoints (LCO 3.4.12)

The power operated relief valves (PORVs) shall each have a nominal maximum lift setting that varies with RCS temperature and which does not exceed the limits in Figure 5.2-8 (Reference 9). The OPPS enable temperature is in accordance with Table 5.2-3. The PORV lift setting provided is for the case with reactor coolant pump (RCP) restrictions. These restrictions are shown in Table 5.2-4, which is taken from Reference 9. Due to the setpoint limitations as a result of the reactor vessel flange requirements, there is no operational benefit achieved by restricting the number of RCPs running to less than two below an indicated RCS temperature of 137°F. Therefore, the PORV setpoints shown in Table 5.2-3 will protect the Appendix G limits for the combinations shown.

The PORV setpoint is based on P/T limits which were established in accordance with 10 CFR 50, Appendix G without allowance for instrumentation error and in accordance with the methodology described in Reference 1, including the exceptions noted in Section 5.2.1. The PORV lift setting shown in Figure 5.2-8 accounts for appropriate instrument error.

5.2.1.3 OPPS Enable Temperature (LCO 3.4.12)

Two different temperatures are used to determine the OPPS enable temperature, they are the arming temperature and the calculated enable temperature. The arming temperature (when the OPPS rendered operable) is established per ASME Section XI, Appendix G. At this temperature, a steam bubble would be present in the pressurizer, thus reducing the potential of a water hammer discharge that could challenge the piping limits. Based on this method, the arming temperature with uncertainty is 237°F.

5.2 Pressure and Temperature Limits Report

The calculated enable temperature is based on either a RCS temperature of less than 200°F or materials concerns (reactor vessel metal temperature less than RT_{NDT} + 50°F), whichever is greater. The calculated enable temperature does not address the piping limit attributed to a water hammer discharge. The calculated enable temperature is 240°F.

As the calculated enable temperature is higher and, therefore, more conservative than the arming temperature, the OPPS enable temperature, as shown in Table 5.2-3, is set to equal the calculated enable temperature.

The calculation method governing the heatup and cooldown of the RCS requires the arming of the OPPS at and below the OPPS enable temperature specified in Table 5.2-3, and disarming of the OPPS above this temperature. The OPPS is required to be enabled, i.e., OPERABLE, when any RCS cold leg temperature is less than or equal to this temperature.

The OPPS enable temperature, PORV setpoints, and RCP operating restrictions contained in Tables 5.2-3 and 5.2-4 and Figure 5.2-8 are as described in Reference 15, and are based upon analysis of Capsule X. The pressure-temperature limits provided in Reference 14 for Capsule X and setpoints evaluation per Reference 15 support the continued use of these existing OPPS/PORV setpoints and RCP operating restrictions for the period up to 30 EFPY. As a result, Tables 5.2-3 and 5.2-4 and Figure 5.2-8 remain valid for Capsule X up to 30 EFPY.

From a plant operations viewpoint the terms "armed" and "enabled" are synonymous when it comes to activating the OPPS. As stated in the applicable operating procedure, the OPPS is activated (armed/enabled) manually before entering the applicability of LCO 3.4.12. This is accomplished by placing two switches (one in each train) into their "ARM" position. Once OPPS is activated (armed/enabled) reactor coolant system pressure transmitters will signal a rise in system pressure above the variable OPPS setpoint. This will initiate an alarm in the control room and open the OPPS PORVs.

5.2.1.4 Reactor Vessel Boltup Temperature (LCO 3.4.3)

The minimum boltup temperature for the Reactor Vessel Flange shall be ≥ 60°F. Boltup is a condition in which the reactor vessel head is installed with tension applied to any stud, and with the RCS vented to atmosphere.

5.2 Pressure and Temperature Limits Report

5.2.2 Reactor Vessel Material Surveillance Program

The reactor vessel material irradiation surveillance specimens shall be removed and analyzed to determine changes in material properties. The capsule withdrawal schedule is provided in Table 5.3-6 of the UFSAR. Also, the results of these analyses shall be used to update Figures 5.2-1 through 5.2-6, and Tables 5.2-1 and 5.2-2 in this report. The time of specimen withdrawal may be modified to coincide with those refueling outages nearest the withdrawal schedule.

The pressure vessel material surveillance program (References 4 and 13) is in compliance with Appendix H to 10 CFR 50, "Reactor Vessel Radiation Surveillance Program." The material test requirements and the acceptance standards utilize the reference nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASME, Section III, NB-2331. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Appendix G, "Protection Against Non-Ductile Failure," to Section XI of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E 185-82.

Reference 10 is an NRC commitment made by FENOC to use only the calculated vessel fluence values when performing future capsule surveillance evaluations for BVPS Unit 2. This commitment is a condition of License Amendment 138 and will remain in effect until the NRC staff approves an alternate methodology to perform these evaluations. Best-estimate values generated using the FERRET Code may be provided for information only.

5.2.3 Supplemental Data Tables

The following tables provide supplemental information on reactor vessel material properties and are provided to be consistent with Generic Letter 96-03. Some of the material property values shown were used as inputs to the P/T limits.

Table 5.2-5, taken from Table 2-4 of Reference 13, shows the calculation of the surveillance material chemistry factors using surveillance capsule data.

Table 5.2-6, taken from Table 2-1 of Reference 14, provides the reactor vessel beltline material property table.

Table 5.2-7, taken from Table 4-2 of Reference 13, provides the reactor vessel extended beltline material property table.

Table 5.2-8, taken from Tables 4-7 and 4-8 of Reference 14, provides a summary of the Adjusted Reference Temperature (ARTs) for 30 EFPY.

5.2 Pressure and Temperature Limits Report

Table 5.2-9, taken from Tables 4-7 and 4-8 of Reference 14, shows the calculation of ARTs for 30 EFPY.

Table 5.2-10, taken from Table 6-3 of Reference 13, provides RT_{PTS} values for the Beltline Region Materials at 54 EFPY.

Table 5.2-11, taken from Table 6-4 of Reference 13, provides RT_{PTS} values for the Extended Beltline Region Materials at 54 EFPY.

Note that Tables 5.2-5, 5.2-8 and 5.2-9 reflect Capsule X analysis and fluence data.

5.2.4 References

1. WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et al., May 2004.

2. (Deleted)

3. (Deleted)

4. WCAP-9615, Revision 1, "Duquesne Light Company, Beaver Valley Unit No. 2 Reactor Vessel Radiation Surveillance Program," P. A. Peter, June 1995.

5. WCAP-15676, "Evaluation of Pressurized Thermal Shock for Beaver Valley Unit 2," J. H. Ledger, August 2001.

6. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," Federal Register, Volume 60, No. 243, December 19, 1995.

7. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," May 15, 1991. (PTS Rule)

8. Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.

9. FENOC Calculation No. 10080-SP-2RCS-006, Revision 4, Addendum 1, "BV-2 LTOPS Setpoint Evaluation Capsule W for 22 EFPY."

10. FirstEnergy Nuclear Operating Company letter L-01-157, "Supplement to License Amendment Requests Nos. 295 and 167," dated December 21, 2001.

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11. (Deleted)

12. WCAP-16527, Revision 0, "Analysis of Capsule X from FirstEnergy Nuclear Operating Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," B. N. Burgos, J. Conermann, S. L. Anderson, March 2006.

13. WCAP-16527, Supplement 1, Revision 1, "Analysis of Capsule X from FirstEnergy Nuclear Operating Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," A. E. Freed, September 2011.

14. WCAP-16528, Revision 1, "Beaver Valley Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," June 2008.

15. Westinghouse Letter FENOC-07-92, dated June 8, 2007, LTOPS Setpoint Evaluation for Beaver Valley Unit 2 Capsule X at 22 and 30 EFPY.

16. Westinghouse Letter MCOE-LTR-13-19, Revision 0, dated March 6, 2013, "Acceptable Initial RT_{NDT} Values for the Beaver Valley Unit 2 Reactor Vessel Inlet Nozzle Materials."

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1
LIMITING ART VALUES AT 30 EFPY: 1/4T, 143°F
3/4T, 132°F
CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 30 EFPY.

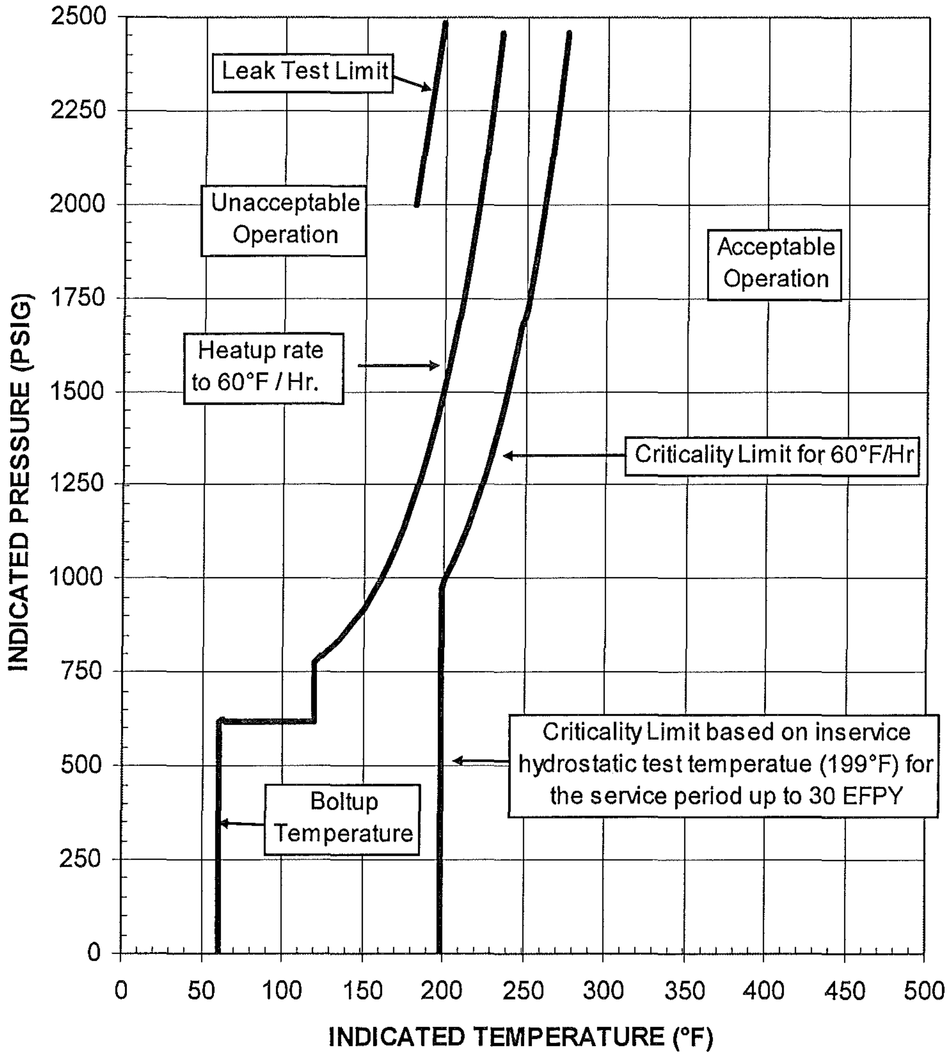

Figure 5.2-1 (Page 1 of 1)
Reactor Coolant System Heatup
Limitations Applicable for the First 30 EFPY (LCO 3.4.3)

<u>MATERIAL PROPERTY BASIS</u>

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1

LIMITING ART VALUES AT 30 EFPY: 1/4T, 143°F
3/4T, 132°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 0°F/HR FOR THE SERVICE PERIOD UP TO 30 EFPY.

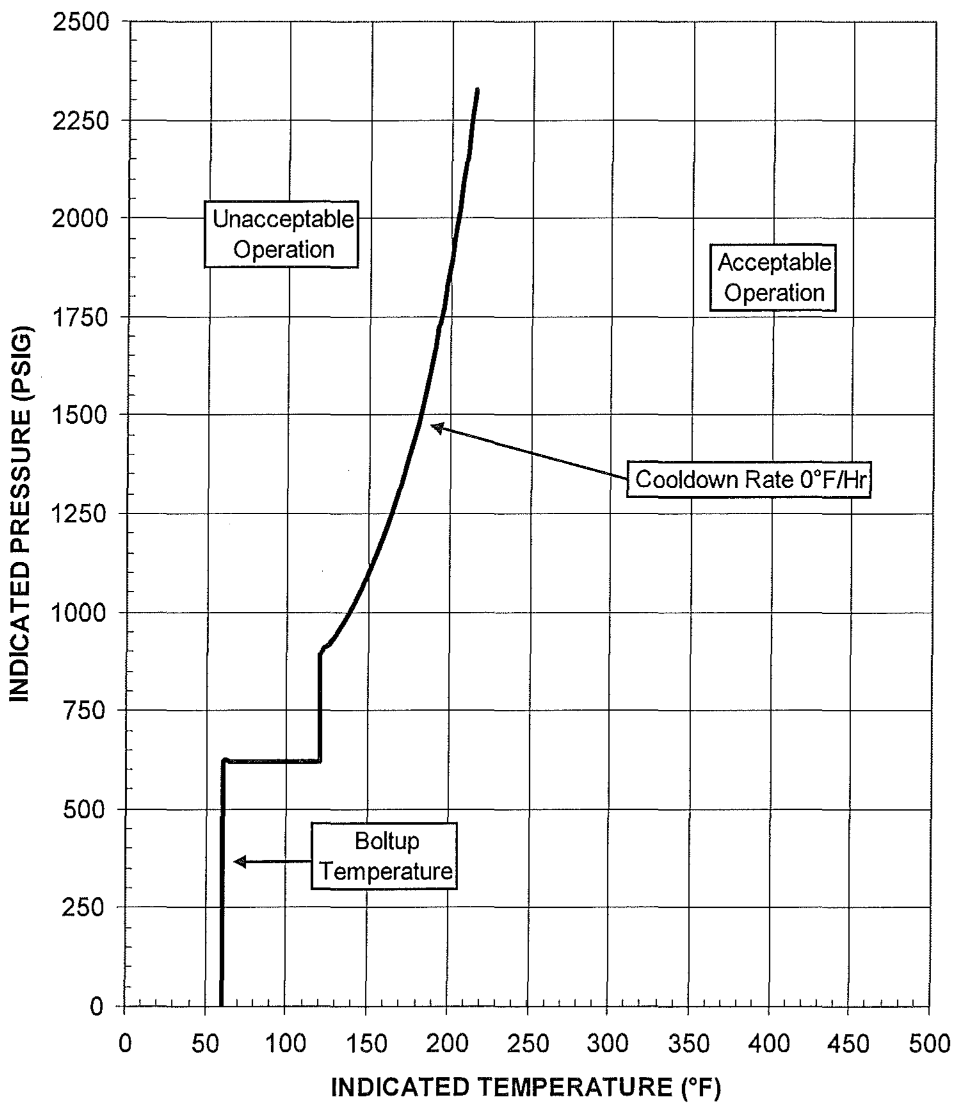

Figure 5.2-2 (Page 1 of 1)
Reactor Coolant System Cooldown (steady state - 0°F/Hr.)
Limitations Applicable for the First 30 EFPY (LCO 3.4.3)

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1
LIMITING ART VALUES AT 30 EFPY: 1/4T, 143°F
3/4T, 132°F
CURVE APPLICABLE FOR COOLDOWN RATES UP TO 20°F/HR FOR THE SERVICE PERIOD UP TO 30 EFPY.

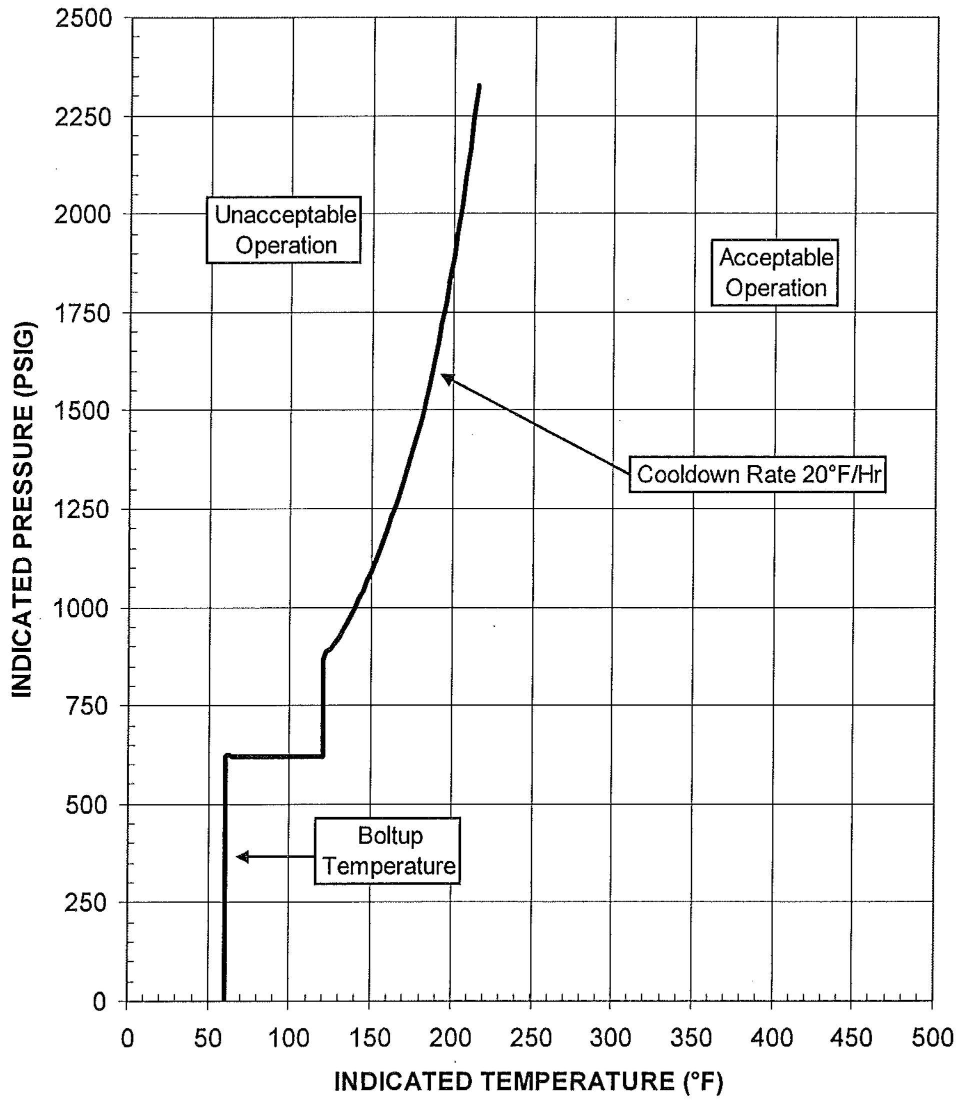

Figure 5.2-3 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 20°F/Hr.)
Limitations Applicable for the First 30 EFPY (LCO 3.4.3)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1

LIMITING ART VALUES AT 30 EFPY: 1/4T, 143°F
3/4T, 132°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 40°F/HR FOR THE SERVICE PERIOD UP TO 30 EFPY.

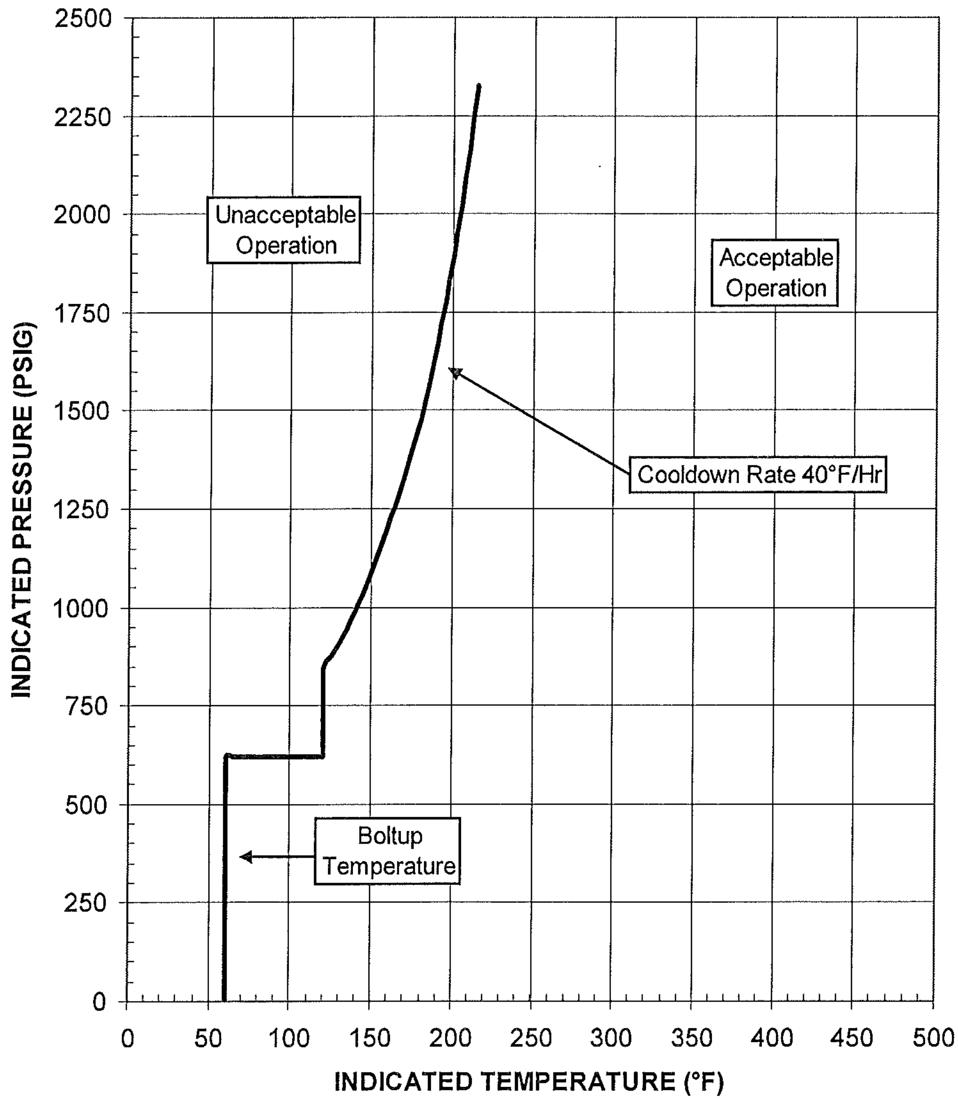

Figure 5.2-4 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 40°F/Hr.)
Limitations Applicable for the First 30 EFPY (LCO 3.4.3)

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1
LIMITING ART VALUES AT 30 EFPY: 1/4T, 143°F
3/4T, 132°F
CURVE APPLICABLE FOR COOLDOWN RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 30 EFPY.

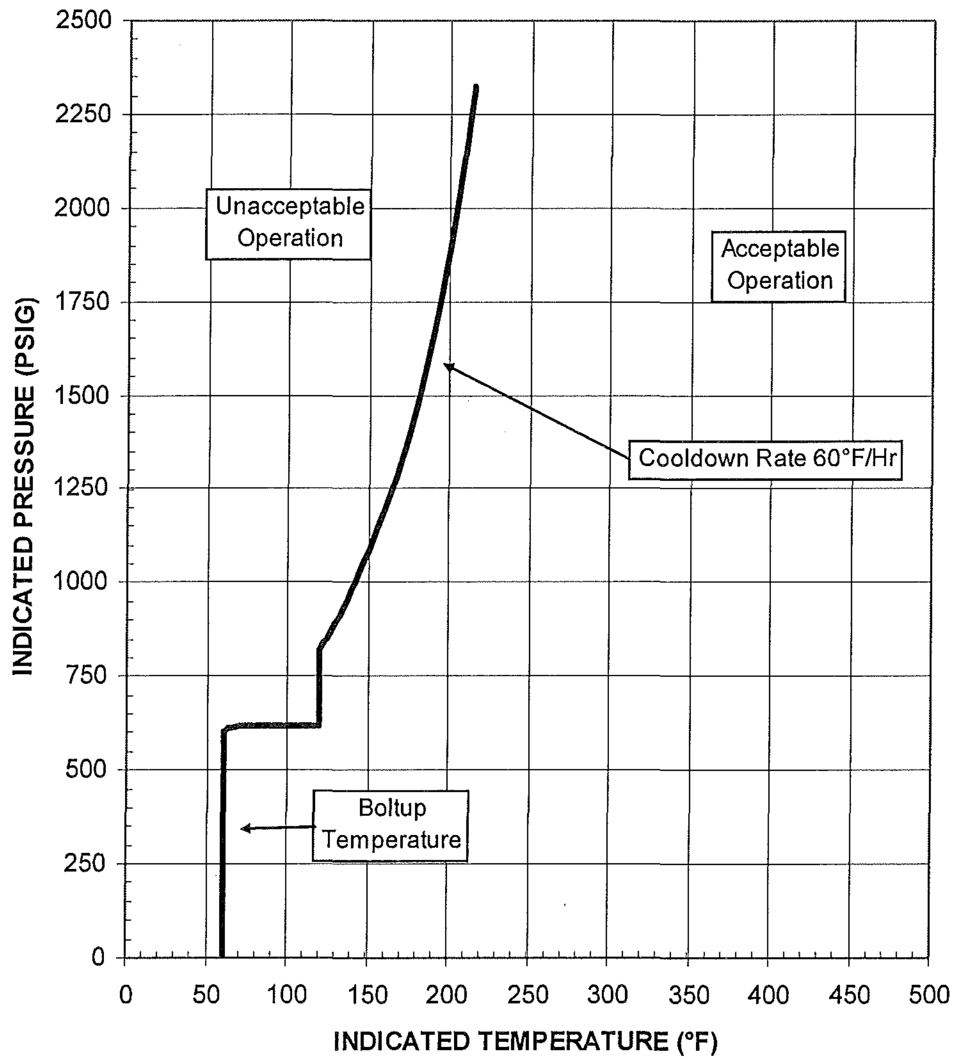

Figure 5.2-5 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 60°F/Hr.)
Limitations Applicable for the First 30 EFPY (LCO 3.4.3)

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1
LIMITING ART VALUES AT 30 EFPY: 1/4T, 143°F
3/4T, 132°F
CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 30 EFPY.

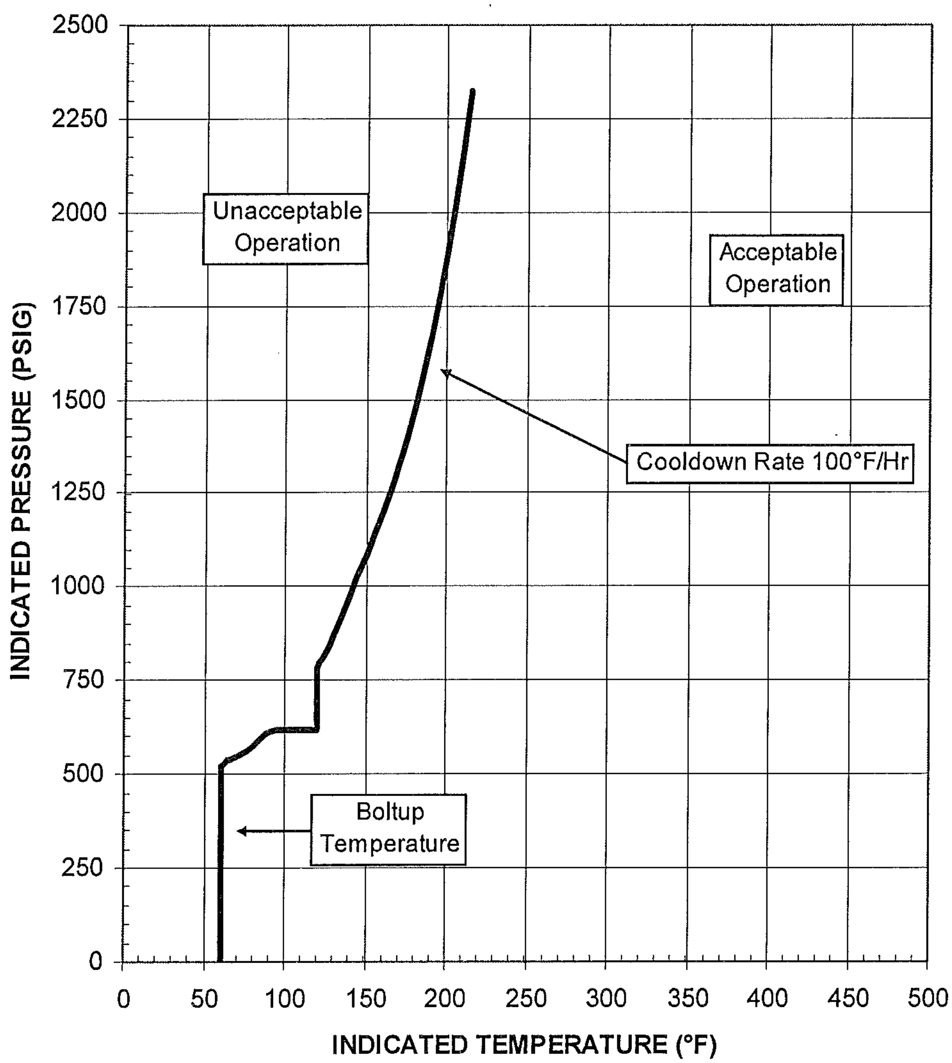

Figure 5.2-6 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 100°F/Hr.)
Limitations Applicable for the First 30 EFPY (LCO 3.4.3)

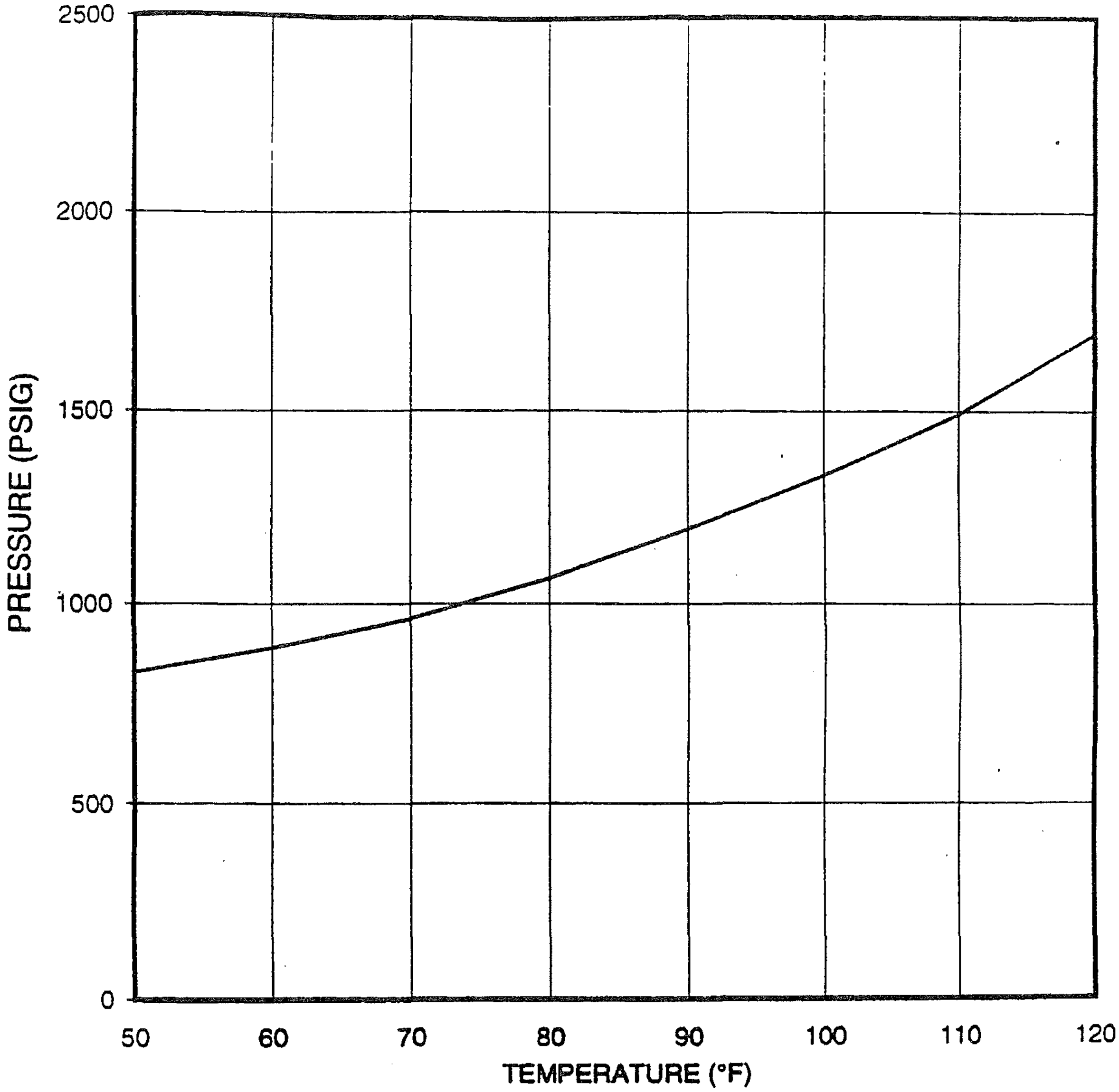

Figure 5.2-7 (Page 1 of 1)
Isolated Loop Pressure – Temperature Limit Curve (LCO 3.4.3)

See Table 5.2-4 for RCP restrictions.

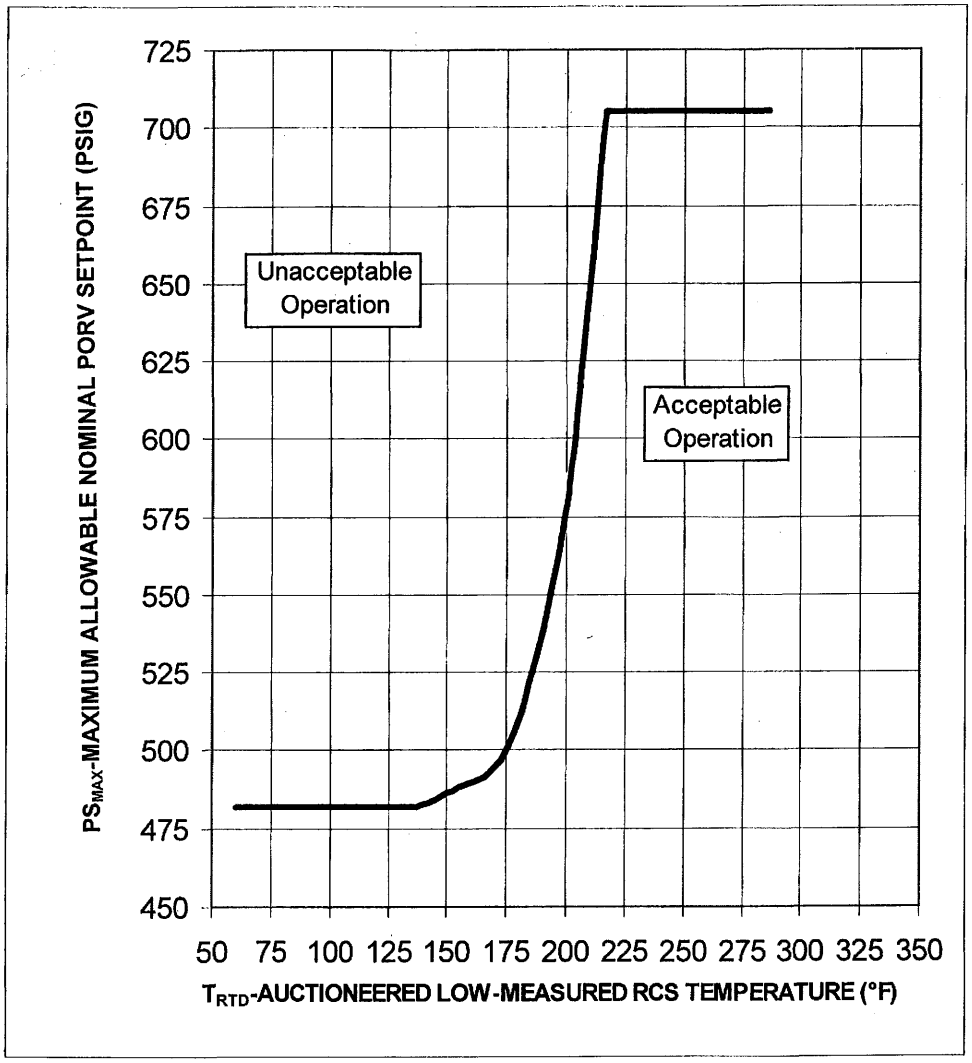

Figure 5.2-8 (Page 1 of 1)
Maximum Allowable Nominal PORV Setpoint for the Overpressure Protection System (LCO 3.4.12)

Table 5.2-1 (Page 1 of 1)
Heatup Curve Data Points for 30 EFPY (LCO 3.4.3)

60°F/HR HEATUP		60°F/HR CRITICALITY		LEAK TEST LIMIT	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	199	0	181	2000
60	621	199	621	199	2485
65	621	199	621		
70	621	199	621		
75	621	199	621		
80	621	199	621		
85	621	199	621		
90	621	199	621		
95	621	199	621		
100	621	199	621		
105	621	199	777		
110	621	199	793		
115	621	199	813		
120	621	199	835		
120	621	199	861		
120	777	199	889		
125	793	199	921		
130	813	199	957		
135	835	200	996		
140	861	205	1040		
145	889	210	1089		
150	921	215	1143		
155	957	220	1203		
160	996	225	1269		
165	1040	230	1342		
170	1089	235	1423		
175	1143	240	1512		
180	1203	245	1611		
185	1269	250	1719		
190	1342	255	1840		
195	1423	260	1972		
200	1512	265	2118		
205	1611	270	2280		
210	1719	275	2458		
215	1840				
220	1972				
225	2118				
230	2280				
235	2458				

Table 5.2-2 (Page 1 of 1)
Cooldown Curve Data Points for 30 EFPY (LCO 3.4.3)

	0°F/HR	20°F/HR	40°F/HR	60°F/HR	100°F/HR
Temp. (°F)	Press. (psig)	Press. (psig)	Press. (psig)	Press. (psig)	Press. (psig)
60	0	0	0	0	0
60	621	621	621	602	525
65	621	621	621	612	536
70	621	621	621	621	548
75	621	621	621	621	562
80	621	621	621	621	578
85	621	621	621	621	595
90	621	621	621	621	614
95	621	621	621	621	621
100	621	621	621	621	621
105	621	621	621	621	621
110	621	621	621	621	621
115	621	621	621	621	621
120	621	621	621	621	621
120	621	621	621	621	621
120	892	867	844	822	783
125	918	896	875	855	823
130	947	927	909	893	867
135	980	962	947	934	917
140	1016	1001	989	980	971
145	1055	1044	1036	1031	1031
150	1099	1092	1087	1087	1087
155	1147	1144	1144	1144	1144
160	1201	1201	1201	1201	1201
165	1260	1260	1260	1260	1260
170	1325	1325	1325	1325	1325
175	1397	1397	1397	1397	1397
180	1477	1477	1477	1477	1477
185	1565	1565	1565	1565	1565
190	1662	1662	1662	1662	1662
195	1770	1770	1770	1770	1770
200	1888	1888	1888	1888	1888
205	2020	2020	2020	2020	2020
210	2165	2165	2165	2165	2165
215	2325	2325	2325	2325	2325

Table 5.2-3 (Page 1 of 1)

Overpressure Protection System (OPPS) Setpoints (LCO 3.4.12)

FUNCTION	SETPOINT
OPPS Enable Temperature	240°F
PORV Setpoint	Figure 5.2-8

Table 5.2-4 (Page 1 of 1)

Reactor Coolant Pump Restrictions

T_{RCS}	Running RCPs
< 137°F	0 – 2
≥ 137°F	3

Table 5.2-5 (Page 1 of 1)
Calculation of Chemistry Factors Using Surveillance Capsule Data

Material	Capsule	Capsule f[(a)]	FF[(b)]	ΔRT_{NDT}[(c)]	FF*ΔRT_{NDT}	FF^2
Intermediate Shell Plate B9004-2[(d)] (Longitudinal)	U	0.615	0.864	24.0	20.73	0.746
	V	2.64	1.260	56.0	70.54	1.587
	W	3.61	1.334	71.0	94.68	1.778
	X	5.63	1.425	98.0	139.65	2.031
Intermediate Shell Plate B9004-2[(d)] (Transverse)	U	0.615	0.864	17.7	15.29	0.746
	V	2.64	1.260	46.1	58.07	1.587
	W	3.61	1.334	63.4	84.55	1.778
	X	5.63	1.425	104.1	148.34	2.031
				SUM:	631.87	12.284
	CF = $\Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2)$ = (631.87) ÷ (12.284) = **51.4°F**					
Beaver Valley Unit 2 Surveillance Weld Metal[(e)] (Heat #83642)	U	0.615	0.864	4.1	3.54	0.746
	V	2.64	1.260	25.7	32.37	1.587
	W	3.61	1.334	6.0	8.00	1.778
	X	5.63	1.425	22.9	32.63	2.031
				SUM:	76.55	6.142
	CF = $\Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2)$ = (76.55) ÷ (6.142) = **12.5°F**					

Notes:

(a) f = calculated surveillance capsule neutron fluence (x 10^{19} n/cm^2, E > 1.0 MeV). The surveillance capsule fluence results are contained in Table 8-1 of Reference 13.

(b) FF = fluence factor = $f^{(0.28 - 0.1 * \log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values. The BVPS-2 ΔRT_{NDT} values for the surveillance weld data were not adjusted since the ratio was 0.91; therefore, a conservative value of 1.00 was used.

(d) The surveillance plate data is deemed non-credible, per Appendix A of Reference 13.

(e) The surveillance weld data is deemed credible, per Appendix A of Reference 13.

Table 5.2-6 (Page 1 of 1)
Reactor Vessel Beltline Material Properties

Material	Cu (wt%)	Ni (wt%)	Initial RT_{NDT} (F)[a]
Closure Head Flange B9002-1	0.06[b]	0.74	-10
Vessel Flange B9001-1	0.06[b]	0.73	0
Intermediate Shell Plate B9004-1	0.065	0.55	60
Intermediate Shell Plate B9004-2	0.06	0.57	40
Lower Shell Plate B9005-1	0.08	0.58	28
Lower Shell Plate B9005-2	0.07	0.57	33
Intermediate to Lower Shell Weld 101-171 (Heat 83642)	0.046	0.086	-30
Intermediate Longitudinal Weld 101-124 A & B (Heat 83642)	0.046	0.086	-30
Lower Longitudinal Weld 101-142 A & B (Heat 83642)	0.046	0.086	-30
Plate Surveillance Material B9004-2	0.06	0.57	40
Surveillance Weld (Heat 83642)	0.065	0.065	-30[c]

Notes:

(a) The initial RT_{NDT} values for all of the beltline materials are based on measured data.

(b) According to the BVPS-2 reactor vessel CMTRs and MISC-PENG-ER-021, the material for the closure head flange (B9002-1) and vessel flange (B9001-1) forgings are ASTM A508 Class 2. The ASTM A508 material specification does not require analysis of copper content. The importance of copper content in the irradiation embrittlement of ferritic pressure vessel steel was not recognized or regulated by the NRC or nuclear steam supply system (NSSS) vendors when the BVPS-2 reactor vessel was constructed. Even though the material specification did not require analysis of copper content for ASTM A508 Class 2 material, check analyses on chemistry measurements (including copper) were reported in MISC-PENGER-021. The copper values reported for both the closure head flange (B9002-1) and the vessel flange (B9001-1) was 0.06%.

(c) The initial RT_{NDT} value is determined in accordance with the requirements of Subparagraph NB-2331 of Section III of the ASME B&PV Code, as specified by Paragraph II - D of 10 CFR Part 50, Appendix G. These fracture toughness requirements are also summarized in Branch Technical Position MTEB Section II.5-2 ("Fracture Toughness") of the NRC Regulatory Standard Review Plan. Following these requirements, along with the Charpy data reported in Table 3-3 of WCAP-9615 and the T_{NDT} value of -30°F defined on page 3-14 of WCAP-9615, the initial RT_{NDT} value is concluded to be equal to T_{NDT} (i.e., -30.0°F).

Table 5.2-7 (Page 1 of 1)
Reactor Vessel Extended Beltline Material Properties [(a)]

Material Description	Material ID	Heat Number	Wt % Cu	Wt% Ni	Initial RT_{NDT} (°F)[(b)]
Upper Shell	B9003-1	A9406-1	0.13	0.60	50
	B9003-2	B4431-2	0.12	0.60	60
	B9003-3	A9406-2	0.13	0.60	50
Upper Shell Longitudinal Welds	101-122A 101-122B 101-122C	51912 (3490)	0.156	0.059	-50
		51912 (3536)	0.156	0.059	-70
		EAIB	0.02	0.98	10 (Gen)
		IAGA	0.03	0.98	-30
		BOHB	0.05	1.00	10 (Gen)
		BAOED	0.02	1.00	-50
Upper Shell to Intermediate Shell Girth Weld	103-121	4P5174 (1122)	0.09	1.00	-50
		51922 (3489)	0.05	1.00	-56 (Gen)
		AAGC	0.03	0.98	-70
		KOIB	0.03	0.97	-60
Inlet Nozzles	B9011-1	2V2436-01-002	0.11	0.85	60[(c)]
	B9011-2	2V2437-02-001	0.13	0.88	60[(c)] (Gen)
	B9011-3	2V2445-02-003	0.13	0.84	70[(c)]
Inlet Nozzle Welds	105-121A 105-121B 105-121C	4P5174 (1122)	0.09	1.00	-50
		LOHB	0.03	1.03	-60
		HABJC	0.02	1.02	-70
		BABBD	0.02	1.04	-70
		FABGC	0.03	1.02	-80
		EOBC	0.02	0.96	-60
		FAAFC	0.07	1.04	-60
		CCJC	0.02	0.99	-60
		FAGB	0.02	1.06	-30
		BAOED	0.02	1.00	-50
Outlet Nozzles	B9012-1	AV8080-2E9558	0.13	0.72	-10
	B9012-2	AV8120-2E9560	0.13	0.74	-10
	B9012-3	AV8097-2E9559	0.13	0.70	-10
Outlet Nozzle Welds	107-121A 107-121B 107-121C	BABBD	0.02	1.04	-70
		FAAFC	0.07	1.04	-60
		HAAEC	0.03	1.03	-80
		HABJC	0.02	1.02	-70
		HAGB	0.02	1.04	-40
		GACJC	0.03	1.00	-80
		JAHB	0.03	0.97	-40

Notes:

(a) Materials information taken from Reference 13
(b) Based on Reference 13, the generic Initial RT_{NDT} values were determined in accordance with NUREG-0800 and the 10 CFR 50.61.
(c) As described in Reference 16, the reactor vessel initial RT_{NDT} values for the inlet nozzles are conservatively assigned values. The actual initial RT_{NDT} values for the reactor vessel inlet nozzles are located in BVPS-2 UFSAR Table 5.3-1.

Table 5.2-8 (Page 1 of 1)

Summary of Adjusted Reference Temperature (ARTs) for 30 EFPY[a]

Material Description	Method Used To Calculate the CF[b]	30 EFPY ART	
		1/4T ART (°F)	3/4T ART (°F)
Intermediate Shell Plate B9004-1	Position 1.1	143	132
Intermediate Shell Plate B9004-2	Position 1.1	119	109
	Position 2.1	119	106
Lower Shell Plate B9005-1	Position 1.1	123	110
Lower Shell Plate B9005-2	Position 1.1	120	109
Vessel Beltline Welds[c]	Position 1.1	53	35
	Position 2.1	0	-6

Notes:

(a) Table reflects Capsule X analysis per Reference 14.

(b) Regulatory Guide 1.99, Revision 2.

(c) All Beltline Welds are from Heat #83642, Linde 0091, Flux Lot #3536.

Table 5.2-9 (Page 1 of 1)

Calculation of Adjusted Reference Temperatures (ARTs) for 30 EFPY[a]

PARAMETER	VALUES	
Operating Time	30 EFPY	
Material – Intermediate Shell Plate	B9004-1	B9004-1
Location	1/4T	3/4T
Chemistry Factor, CF (°F)	40.5	40.5
Fluence, (f), (10^{19} n/cm^2)[b]	2.113	0.8215
Fluence Factor, FF	1.203	0.9448
ΔRT_{NDT} = CF x FF(°F)	48.74	38.27
Intitial RT_{NDT}, I(°F)	60	60
Margin, M(°F)	34	34
ART, per Regulatory Guide 1.99, Revision 2	143	132

Notes:

(a) Table reflects Capsule X analysis per Reference 14.

(b) Fluence (f), is based upon f_{surf} (10^{19} n/cm^2, E > 1.0 MeV) = 3.39 at 30 EFPY. The Beaver Valley Unit 2 reactor vessel wall thickness is 7.875 inches at the beltline region.

Table 5.2-10 (Page 1 of 1)
RT_{PTS} Calculation for Beltline Region Materials at Life Extension (54 EFPY)[a]

Material Description	Material ID	Heat Number	Surface Neutron Fluence ($\times 10^{19}$ n/cm^2)	Fluence Factor, FF[b]	Chemistry Factor (°F)	Initial RT_{NDT}[c] (°F)	ΔRT_{PTS}[d] (°F)	σ_U (°F)	σ_Δ (°F)	Margin[e] (°F)	RT_{PTS}[f] (°F)
Intermediate Shell Plate	B9004-1	- - -	5.18	1.4092	40.5	60	57.1	0	17	34	151.1
Intermediate Shell Plate	B9004-2	- - -	5.18	1.4092	37	40	52.1	0	17	34	126.1
→ Using non-credible surveillance data[g]			5.18	1.4092	51.4	40	72.4	0	17	34	146.4
Lower Shell Plate	B9005-1	- - -	5.21	1.4104	51	28	71.9	0	17	34	133.9
Lower Shell Plate	B9005-2	- - -	5.21	1.4104	44	33	62.1	0	17	34	129.1
Intermediate to Lower Shell Girth Weld	101-171	83642	5.18	1.4092	34.4	-30	48.5	0	24.2	48.5	67.0
→ Using credible surveillance data[g]			5.18	1.4092	12.5	-30	17.6	0	8.8	17.6	5.2
Intermediate Shell Longitudinal Welds	101-124 A&B	83642	1.76	1.1554	34.4	-30	39.7	0	19.9	39.7	49.5
→ Using credible surveillance data[g]			1.76	1.1554	12.5	-30	14.4	0	7.2	14.4	-1.1
Lower Shell Longitudinal Welds	101-142 A&B	83642	1.77	1.1569	34.4	-30	39.8	0	19.9	39.8	49.6
→ Using credible surveillance data[g]			1.77	1.1569	12.5	-30	14.5	0	7.2	14.5	-1.1

Notes:
(a) Data obtained from Table 6-3 of Reference 13.
(b) FF = fluence factor = $f^{(0.28 - 0.1 \log (f))}$.
(c) Initial RT_{NDT} values are measured values.
(d) ΔRT_{PTS} = CF * FF.
(e) $M = 2 * (\sigma_U^2 + \sigma_\Delta^2)^{1/2}$.
(f) RT_{PTS} = Initial RT_{NDT} + ΔRT_{PTS} + Margin.
(g) The BVPS-2 surveillance weld metal is the same weld heat as the BVPS-2 beltline welds (heat 83642). The BVPS-2 surveillance weld data is credible; therefore, the reduced σ_Δ term of 14°F was utilized for BVPS-2 weld heat 83642. The BVPS-2 surveillance plate material is representative of the BVPS-2 intermediate shell plate B9004-2. The surveillance plate material is non-credible; therefore, the higher σ_Δ term of 17°F was utilized for BVPS-2 plate B9004-2. The credibility evaluation conclusions are contained in Appendix A of Reference 13.

Table 5.2-11 (Page 1 of 3)
RT_{PTS} Calculation for Extended Beltline Region Materials at Life Extension (54 EFPY)[a]

Material Description	Material ID	Heat Number (Lot Number)	Surface Neutron Fluence ($x10^{19}$ n/cm^2)	Fluence Factor, FF[b]	Chemistry Factor (°F)	Initial RT_{NDT}[c] (°F)	ΔRT_{PTS}[e] (°F)	σ_U (°F)	σ_Δ (°F)	Margin[f] (°F)	RT_{PTS}[g] (°F)
Upper Shell Plates	B9003-1	A9406-1	0.515	0.8147	91.0	50	74.1	0	17	34	158.1
	B9003-2	B4431-2	0.515	0.8147	83.0	60	67.6	0	17	34	161.6
	B9003-3	A9406-2	0.515	0.8147	91.0	50	74.1	0	17	34	158.1
Upper Shell Longitudinal Welds	101-122A 101-122B 101-122C	51912 (3490)	0.515	0.8147	73.71	-50	60.1	0	28	56	66.1
		51912 (3536)	0.515	0.8147	73.71	-70	60.1	0	28	56	46.1
		EAIB	0.515	0.8147	27.0	10[d]	22.0	17	11.0	40.5	72.5
		IAGA	0.515	0.8147	41.0	-30	33.4	0	16.7	33.4	36.8
		BOHB	0.515	0.8147	68.0	10[d]	55.4	17	27.7	65.0	130.4
		BAOED	0.515	0.8147	27.0	-50	22.0	0	11.0	22.0	-6.0
Upper to Intermediate Shell Girth Weld	103-121	4P5174	0.515	0.8147	122.0	-50	99.4	0	28	56.0	105.4
		51922	0.515	0.8147	68.0	-56[d]	55.4	17	27.7	65.0	64.4
		AAGC	0.515	0.8147	41.0	-70	33.4	0	16.7	33.4	-3.2
		KOIB	0.515	0.8147	41.0	-60	33.4	0	16.7	33.4	6.8
Inlet Nozzles	B9011-1	2V2436-01-002	0.0298	0.2188	77.0	60[h]	16.8	0	8.4	16.8	93.7
	B9011-2	2V2437-02-001	0.0298	0.2188	96.0	60[d][h]	21.0	17	10.5	40.0	121.0
	B9011-3	2V2445-02-003	0.0298	0.2188	96.0	70[h]	21.0	0	10.5	21.0	112.0

Table 5.2-11 (Page 2 of 3)
RT_{PTS} Calculation for Extended Beltline Region Materials at Life Extension (54 EFPY)[(a)]

Material Description	Material ID	Heat Number (Lot Number)	Surface Neutron Fluence ($x10^{19}$ n/cm^2)	Fluence Factor, FF[(b)]	Chemistry Factor (°F)	Initial RT_{NDT}[(c)] (°F)	ΔRT_{PTS}[(e)] (°F)	σ_U (°F)	σ_Δ (°F)	Margin[(f)] (°F)	RT_{PTS}[(g)] (°F)
Inlet Nozzle Welds	105-121A 105-121B 105-121C	4P5174	0.0298	0.2188	122.0	-50	26.7	0	13.3	26.7	3.4
		LOHB	0.0298	0.2188	41.0	-60	9.0	0	4.5	9.0	-42.1
		HABJC	0.0298	0.2188	27.0	-70	5.9	0	3.0	5.9	-58.2
		BABBD	0.0298	0.2188	27.0	-70	5.9	0	3.0	5.9	-58.2
		FABGC	0.0298	0.2188	41.0	-80	9.0	0	4.5	9.0	-62.1
		EOBC	0.0298	0.2188	27.0	-60	5.9	0	3.0	5.9	-48.2
		FAAFC	0.0298	0.2188	95.0	-60	20.8	0	10.4	20.8	-18.4
		CCJC	0.0298	0.2188	27.0	-60	5.9	0	3.0	5.9	-48.2
		FAGB	0.0298	0.2188	27.0	-30	5.9	0	3.0	5.9	-18.2
		BAOED	0.0298	0.2188	27.0	-50	5.9	0	3.0	5.9	-38.2
Outlet Nozzles	B9012-1	AV8080-2E9558	0.0151	0.1440	94.0	-10	13.5	0	6.8	13.5	17.1
	B9012-2	AV8120-2E9560	0.0151	0.1440	94.5	-10	13.6	0	6.8	13.6	17.2
	B9012-3	AV8097-2E9559	0.0151	0.1440	93.5	-10	13.5	0	6.7	13.5	16.9
Outlet Nozzle Welds	107-121A 107-121B 107-121C	BABBD	0.0151	0.1440	27.0	-70	3.9	0	1.9	3.9	-62.2
		FAAFC	0.0151	0.1440	95.0	-60	13.7	0	6.8	13.7	-32.6
		HAAEC	0.0151	0.1440	41.0	-80	5.9	0	3.0	5.9	-68.2
		HABJC	0.0151	0.1440	27.0	-70	3.9	0	1.9	3.9	-62.2
		HAGB	0.0151	0.1440	27.0	-40	3.9	0	1.9	3.9	-32.2
		GACJC	0.0151	0.1440	41.0	-80	5.9	0	3.0	5.9	-68.2
		JAHB	0.0151	0.1440	41.0	-40	5.9	0	3.0	5.9	-28.2

Table 5.2-11 (Page 3 of 3)
RT_{PTS} Calculation for Extended Beltline Region Materials at Life Extension (54 EFPY)[a]

Notes:

(a) Data obtained from Table 6-4 of Reference 13.

(b) FF = fluence factor = $f^{(0.28 - 0.1 \log (f))}$.

(c) Initial RT_{NDT} values are measured values, unless otherwise noted.

(d) Initial RT_{NDT} values are generic.

(e) ΔRT_{PTS} = CF * FF.

(f) $M = 2 * (\sigma_U^2 + \sigma_\Delta^2)^{1/2}$.

(g) RT_{PTS} = Initial RT_{NDT} + ΔRT_{PTS} + Margin.

(h) As described in Reference 16, the reactor vessel initial RT_{NDT} values for the inlet nozzles are conservatively assigned values. The actual initial RT_{NDT} values for the reactor vessel inlet nozzles are located in BVPS-2 UFSAR Table 5.3-1.