Energy Harbor Nuclear Corp.
Beaver Valley Power Station
P. O. Box 4
Shippingport, PA 15077

John J. Grabnar
Site Vice President, Beaver Valley Nuclear

724-682-5234

July 6, 2022
L-22-154

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 5.6.4, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," Energy Harbor Nuclear Corp. hereby submits the BVPS-2 PTLR, Revision 8. Technical Specification 5.6.4.c requires that the PTLR be provided to the Nuclear Regulatory Commission (NRC) upon issuance for any revision or supplement thereto.

Revision 8 of the BVPS-2 PTLR was made effective on June 9, 2022 and updated to incorporate the pressure and temperature limit curves and over-pressure protection system setpoints that are valid through 54 effective full power years of BVPS-2 operation.

The BVPS-2 PTLR, Revision 8, is enclosed.

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,

John J
Grabnar

Digitally signed by John J Grabnar
Date: 2022.07.06 13:37:19 -04'00'

John J. Grabnar

Enclosure: Beaver Valley Power Station Unit No. 2,
Pressure and Temperature Limits Report, Revision 8

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

Enclosure
L-22-154

Beaver Valley Power Station Unit No. 2,
Pressure and Temperature Limits Report, Revision 8

(27 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	5.2-1 5.2-2
3.4.6	5.2.1.3	N/A	N/A
3.4.7	5.2.1.3	N/A	N/A
3.4.10	5.2.1.3	N/A	N/A
3.4.12	5.2.1.2 5.2.1.3	5.2-4	5.2-3 5.2-4
3.5.2	5.2.1.3	N/A	N/A

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 methodologies listed in References 1 and 2 were used.

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 100°F in any one hour period.

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

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.

5.2 Pressure and Temperature Limits Report

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 Figure 5.2-2 and Table 5.2-2. These limits are defined in Reference 14, and are consistent with the methodology described in References 1 and 2. The RCS P/T limits for heatup and cooldown shown in Figures 5.2-1 and 5.2-2 are provided without margins for instrument error. The criticality limit curves in Figure 5.2-1 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 5.2-3 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.

- NOTE -

Pressure limits are considered to be met for pressures that are below 0 psig (i.e., up to and including full vacuum conditions) since the resulting P/T combination is located in the region to the right and below the operating limits provided in Figures 5.2-1, 5.2-2, and 5.2-3.

Reference 12 provides an updated surveillance capsule credibility evaluation, updated Position 2.1 chemistry factor values and an updated fluence evaluation. Therefore, the P/T limit curves provided in Reference 14 were created 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, the limiting material for the BVPS-2 54 EFPY P/T limits is the intermediate shell plate B9004-2.

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

The OPPS 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 5.2-4 and Table 5.2-3. The OPPS enable temperature is defined in Section 5.2.1.3. The PORV lift settings require the reactor coolant pump (RCP) restrictions specified in Table 5.2-4 to be followed.

The OPPS PORV setpoints were calculated in Reference 15 based on the Reference 14 P/T limits in accordance with 10 CFR 50, Appendix G and the methodology described in Reference 1. The OPPS enable temperature in Section 5.2.1.3, PORV setpoints in Figure 5.2-4 and Table 5.2-3, and RCP operating restrictions contained in Table 5.2-4 account for appropriate instrument error. The P/T limits provided in Reference 14 for Capsule Y and setpoints evaluation per Reference 15 support the use of these OPPS PORV setpoints, enable temperature, and RCP operating restrictions for the period up to 54 EFPY. As a result, Tables 5.2-3 and 5.2-4and Figure 5.2-4 remain valid for up to 54 EFPY.

5.2.1.3 OPPS Enable Temperature (LCO 3.4.12)

The OPPS arming/enable temperature (when the OPPS rendered operable) is established per ASME Code Case N-641 based on calculations in Reference 14. The OPPS enable temperature was calculated to be 226°F, which includes margin for the temperature uncertainty. At temperatures above the enable 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. Therefore, the OPPS does not need to be armed at temperatures greater than 226°F for reactor vessel overpressure protection or for pressurizer PORV discharge piping protection.

5.2 Pressure and Temperature Limits Report

The calculation method governing the heatup and cooldown of the RCS requires the arming of the OPPS at and below the OPPS enable temperature of 226°F. The OPPS is required to be enabled, i.e., OPERABLE, when any RCS cold leg temperature is less than or equal to this temperature.

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 and 5.2-2, 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 12) 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 5-1 of Reference 14, shows the calculation of the surveillance material chemistry factors using surveillance capsule data.

Table 5.2-6, taken from Tables 3-1 and 3-2 of Reference 14, provides the reactor vessel beltline material property table.

Table 5.2-7, taken from Table 3-1 of Reference 14, provides the reactor vessel extended beltline material property table.

Table 5.2-8, taken from Tables 7-2 and 7-3 of Reference 14, provides a summary of the Adjusted Reference Temperature (ARTs) for 54 EFPY.

5.2 Pressure and Temperature Limits Report

Table 5.2-9, taken from Tables 7-2 and 7-3 of Reference 14, shows the calculation of the limiting ARTs for 54 EFPY.

Table 5.2-10, taken from Table D-1 of Reference 14, provides RT_{PTS} values for the Beltline Region Materials at 54 EFPY.

Table 5.2-11, taken from Table D-1 of Reference 14, provides RT_{PTS} values for the Extended Beltline Region Materials at 54 EFPY.

Note that Table 5.2-5 and Table 5.2-8 through 5.2-11 reflect Capsule Y 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. WCAP-18124-NP-A, Revision 0, Fluence Determination with RAPTOR-M3G and FERRET,” G. A. Fischer, et al., July 2018.

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. (Deleted)

6. 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements.”

7. 10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.” (PTS Rule)

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

9. (Deleted)

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

5.2 Pressure and Temperature Limits Report

11. (Deleted)

12. WCAP-18558-NP, Revision 0, "Analysis of Capsule Y from the Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," June 2020.

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-18559-NP, Revision 2, "Beaver Valley Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and Supplemental Reactor Vessel Integrity Evaluations Based on Capsule Y Data," July 2021.

15. Westinghouse Letter EH-21-026, Revision 0, “Transmittal of Final Beaver Valley Unit 2 Overpressure Protection System Analysis,” June 22, 2021.

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 B9004-2
LIMITING ART VALUES AT 54 EFPY: 1/4T, 158°F
3/4T, 143°F
CURVES APPLICABLE FOR HEATUP RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 54 EFPY WITHOUT INSTRUMENT ERROR.

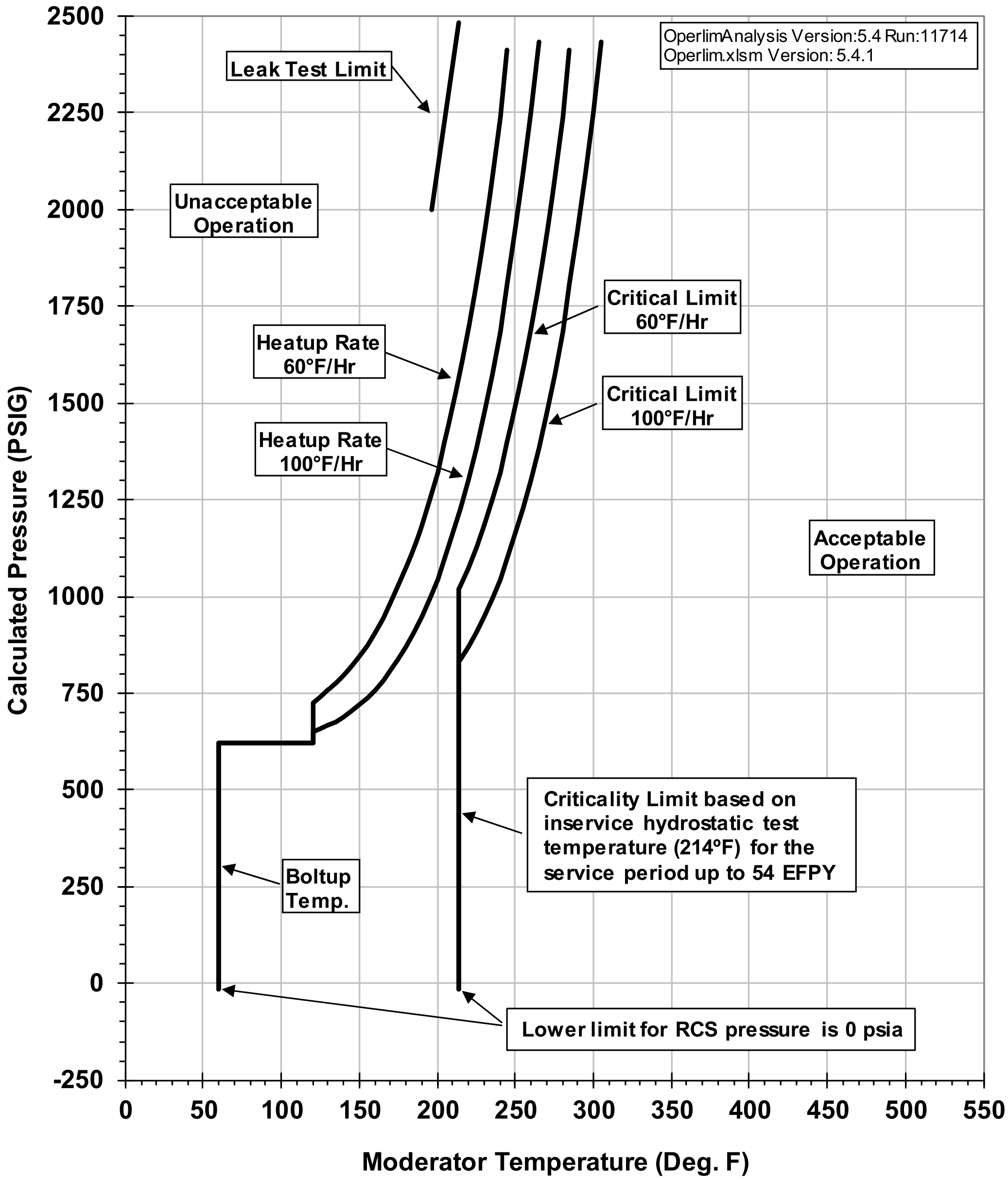

Figure 5.2-1 (Page 1 of 1)
Reactor Coolant System Heatup
Limitations Applicable for the First 54 EFPY (LCO 3.4.3)

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B9004-2
LIMITING ART VALUES AT 54 EFPY: 1/4T, 158°F
3/4T, 143°F
CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 54 EFPY WITHOUT INSTRUMENT ERROR.

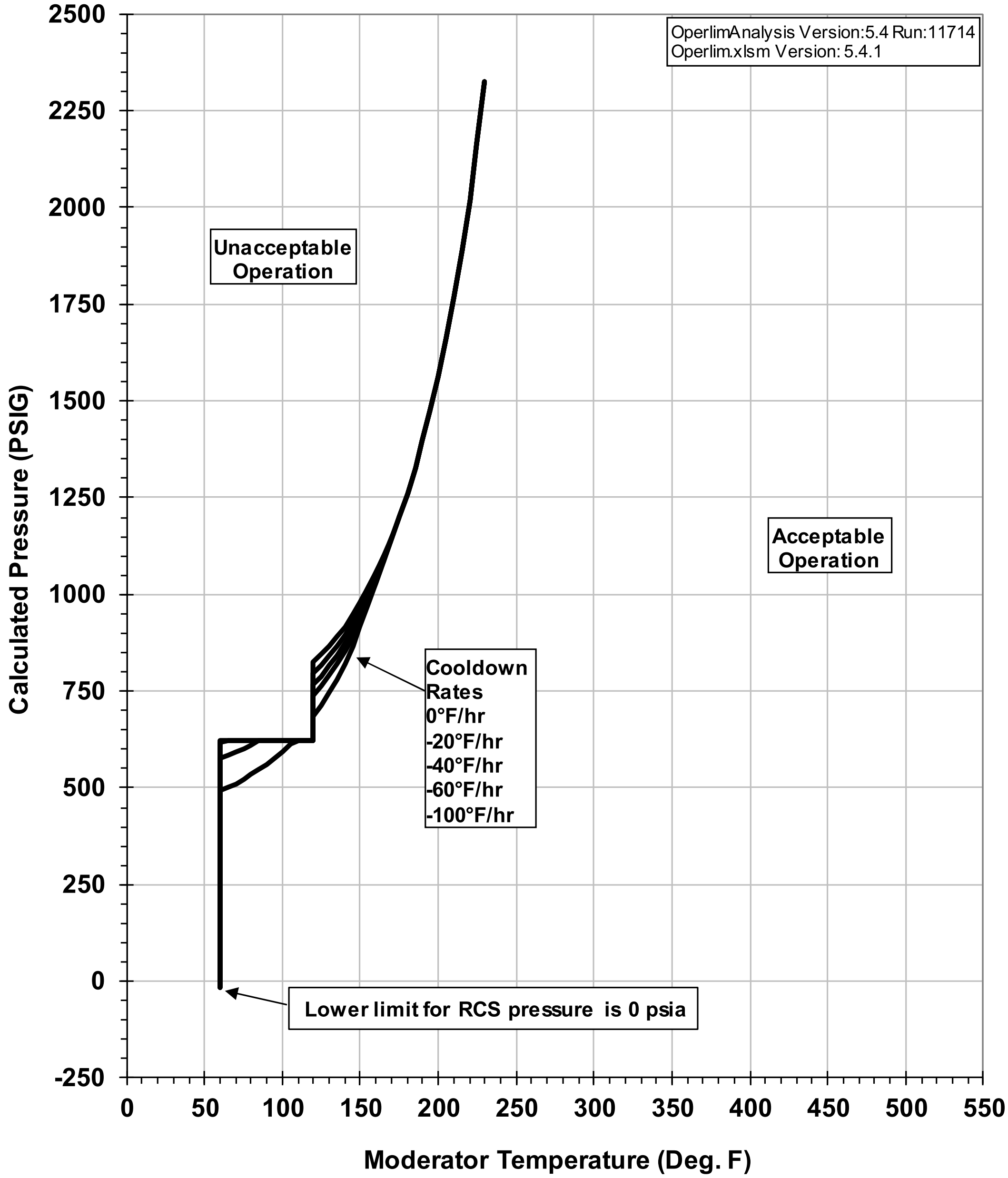

Figure 5.2-2 (Page 1 of 1)
Reactor Coolant System Cooldown (steady state - 0°F/Hr. through 100°F/Hr.)
Limitations Applicable for the First 54 EFPY (LCO 3.4.3)

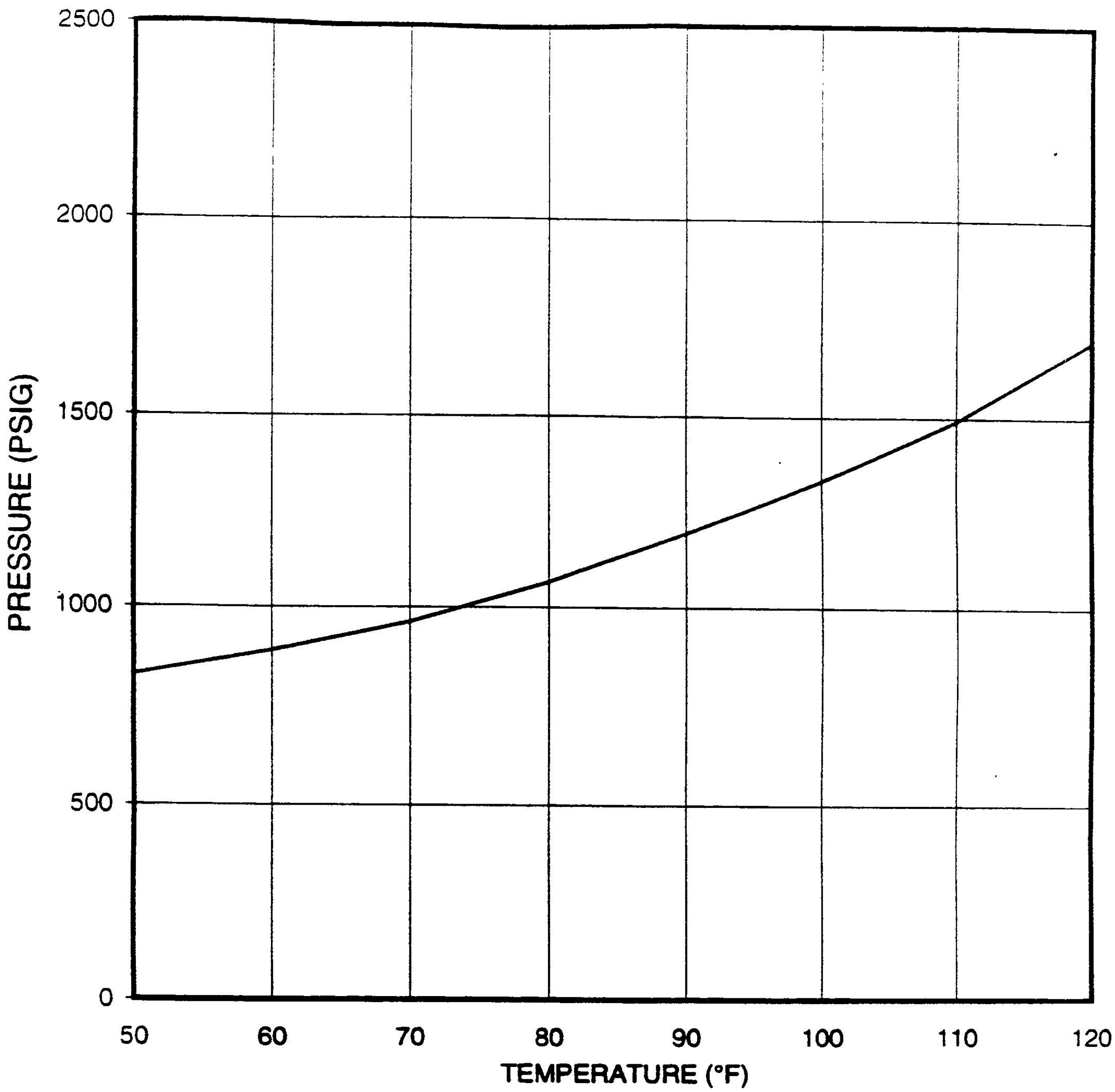

Figure 5.2-3 (Page 1 of 1)
Isolated Loop Pressure – Temperature Limit Curve (LCO 3.4.3)

See Table 5.2-3 for maximum allowable OPPS PORV setpoints and Table 5.2-4 for RCP restrictions.

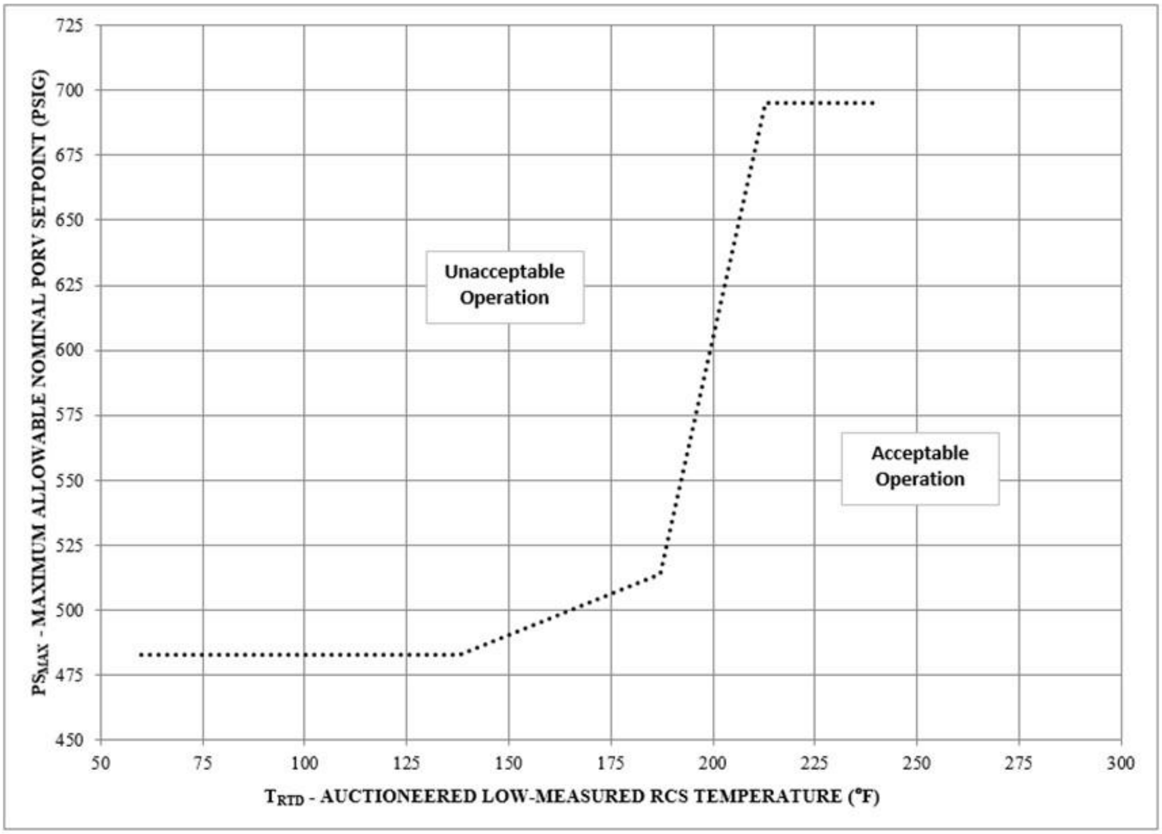

Figure 5.2-4 (Page 1 of 1)
Maximum Allowable Nominal PORV Setpoint for the
Overpressure Protection System with Instrument Error (LCO 3.4.12)

Table 5.2-1 (Page 1 of 2)
Heatup Curve Data Points for 54 EFPY without Instrument Error (LCO 3.4.3)

60°F/hr Heatup		60°F/hr Criticality		100°F/hr Heatup		100°F/hr Criticality	
T (°F)	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**
60	-14.7	214	-14.7	60	-14.7	214	-14.7
60	621	214	1017	60	621	214	833
65	621	215	1025	65	621	215	839
70	621	220	1073	70	621	220	872
75	621	225	1127	75	621	225	908
80	621	230	1185	80	621	230	949
85	621	235	1250	85	621	235	994
90	621	240	1322	90	621	240	1044
95	621	245	1401	95	621	245	1099
100	621	250	1488	100	621	250	1160
105	621	255	1584	105	621	255	1227
110	621	260	1691	110	621	260	1301
115	621	265	1808	115	621	265	1384
120	621	270	1938	120	621	270	1474
120	726	275	2081	120	651	275	1574
125	741	280	2238	125	658	280	1685
130	757	285	2413	130	666	285	1807
135	776	-	-	135	677	290	1941
140	797	-	-	140	689	295	2090
145	820	-	-	145	703	300	2253
150	846	-	-	150	719	305	2434
155	875	-	-	155	738	-	-
160	907	-	-	160	759	-	-
165	943	-	-	165	783	-	-
170	982	-	-	170	809	-	-
175	1025	-	-	175	839	-	-
180	1073	-	-	180	872	-	-
185	1127	-	-	185	908	-	-
190	1185	-	-	190	949	-	-
195	1250	-	-	195	994	-	-
200	1322	-	-	200	1044	-	-
205	1401	-	-	205	1099	-	-
210	1488	-	-	210	1160	-	-

Table 5.2-1 (Page 2 of 2)
Heatup Curve Data Points for 54 EFPY without Instrument Error (LCO 3.4.3)

60°F/hr Heatup		60°F/hr Criticality		100°F/hr Heatup		100°F/hr Criticality	
T (°F)	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**
215	1584	-	-	215	1227	-	-
220	1691	-	-	220	1301	-	-
225	1808	-	-	225	1384	-	-
230	1938	-	-	230	1474	-	-
235	2081	-	-	235	1574	-	-
240	2238	-	-	240	1685	-	-
245	2413	-	-	245	1807	-	-
-	-	-	-	250	1941	-	-
-	-	-	-	255	2090	-	-
-	-	-	-	260	2253	-	-
-	-	-	-	265	2434	-	-

Leak Test Limit	
T (°F)	**P (psig)**
196	2000
214	2485

Table 5.2-2 (Page 1 of 2)
Cooldown Curve Data Points for 54 EFPY without Instrument Error (LCO 3.4.3)

Steady-State		**20°F/hr Cooldown**		**40°F/hr Cooldown**		**60°F/hr Cooldown**		**100°F/hr Cooldown**	
T (°F)	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**
60	-14.7	60	-14.7	60	-14.7	60	-14.7	60	-14.7
60	621	60	621	60	617	60	577	60	495
65	621	65	621	65	621	65	584	65	503
70	621	70	621	70	621	70	592	70	512
75	621	75	621	75	621	75	600	75	522
80	621	80	621	80	621	80	610	80	533
85	621	85	621	85	621	85	621	85	546
90	621	90	621	90	621	90	621	90	560
95	621	95	621	95	621	95	621	95	575
100	621	100	621	100	621	100	621	100	592
105	621	105	621	105	621	105	621	105	612
110	621	110	621	110	621	110	621	110	621
115	621	115	621	115	621	115	621	115	621
120	621	120	621	120	621	120	621	120	621
120	826	120	796	120	767	120	738	120	683
125	846	125	817	125	789	125	763	125	713
130	868	130	841	130	815	130	790	130	745
135	892	135	867	135	843	135	821	135	781
140	918	140	895	140	874	140	854	140	822
145	947	145	927	145	908	145	892	145	866
150	980	150	962	150	946	150	933	150	916
155	1016	155	1001	155	989	155	979	155	971
160	1055	160	1044	160	1035	160	1030	160	1030
165	1099	165	1091	165	1087	165	1086	165	1086
170	1147	170	1144	170	1144	170	1144	170	1144
175	1201	175	1201	175	1201	175	1201	175	1201
180	1260	180	1260	180	1260	180	1260	180	1260
185	1325	185	1325	185	1325	185	1325	185	1325
190	1397	190	1397	190	1397	190	1397	190	1397

Table 5.2-2 (Page 2 of 2)
Cooldown Curve Data Points for 54 EFPY without Instrument Error (LCO 3.4.3)

Steady-State		20°F/hr Cooldown		40°F/hr Cooldown		60°F/hr Cooldown		100°F/hr Cooldown	
T (°F)	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**
195	1477	195	1477	195	1477	195	1477	195	1477
200	1565	200	1565	200	1565	200	1565	200	1565
205	1662	205	1662	205	1662	205	1662	205	1662
210	1769	210	1769	210	1769	210	1769	210	1769
215	1888	215	1888	215	1888	215	1888	215	1888
220	2020	220	2020	220	2020	220	2020	220	2020
225	2165	225	2165	225	2165	225	2165	225	2165
230	2325	230	2325	230	2325	230	2325	230	2325

Table 5.2-3 (Page 1 of 1)
Overpressure Protection System (OPPS) Setpoints with Instrument Error (LCO 3.4.12)

Indicated RCS Temperature (°F)	Maximum Allowable PORV Setpoint (psig)
60	483
138	483
187	514
213	695
226	695
240	695

Notes:

1. The OPPS PORV setpoint is maintained up to a temperature of 240°F, which is greater than the enable temperature of 226°F described in Section 5.2.1.3. As described in Section 5.2.1.3, OPPS is not required above 226°F. Maintaining the OPPS PORV setpoint to 240°F is conservative, but not required for protection.
2. At temperatures greater than 240°F, the nominal OPPS settings implemented in the plant increase to the nominal PORV setting of 2335 psig to preclude PORV opening if the OPPS is inadvertently armed.

Table 5.2-4 (Page 1 of 1)

Reactor Coolant Pump Restrictions with Instrument Error

T_{RCS}	Running RCPs
≤ 137ºF	0 – 2
> 137ºF	3

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

Material	**Capsule**	**Capsule Fluence[a] (x 10^{19} n/cm^2, E > 1.0 MeV)**	**FF[b]**	**ΔRT_{NDT}[c] (°F)**	**FF*ΔRT_{NDT} (°F)**	**FF^2**
Intermediate Shell Plate B9004-2[d] (Longitudinal)	U	0.610	0.862	24.0	20.68	0.742
	V	2.62	1.258	56.0	70.44	1.582
	W	3.68	1.338	71.8	96.06	1.790
	X	5.58	1.423	98.0	139.49	2.026
	Y	8.97	1.500	143.2	214.76	2.249
Intermediate Shell Plate B9004-2[d] (Transverse)	U	0.610	0.862	18.2	15.68	0.742
	V	2.62	1.258	46.5	58.49	1.582
	W	3.68	1.338	63.8	85.36	1.790
	X	5.58	1.423	104.5	148.74	2.026
	Y	8.97	1.500	123.5	185.21	2.249
				SUM:	1034.90	16.779
	$CF_{B9004\text{-}2} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (1034.90) \div (16.779) =$ **61.7°F**					
Beaver Valley Unit 2 Surveillance Weld[e] (Heat #83642)	U	0.610	0.862	5.8	5.00	0.742
	V	2.62	1.258	26.6	33.46	1.582
	W	3.68	1.338	6.0	8.03	1.790
	X	5.58	1.423	23.4	33.31	2.026
	Y	8.97	1.500	36.6	54.89	2.249
				SUM:	134.68	8.389
	$CF_{Surv.\ Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (134.68) \div (8.389) =$ **16.1°F**					

<u>Notes:</u>

(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 5-1 of Reference 14.

(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 0.905; therefore, a conservative value of 1.00 was used.

(d) The surveillance plate data is deemed non-credible, per Appendix D of Reference 12.

(e) The surveillance weld data is deemed credible, per Appendix D of Reference 12.

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[(d)]	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[(d)]	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 14, unless otherwise noted.
(b) Based on Reference 14, 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.
(d) The material properties were taken from Reference 13. The outlet nozzles and outlet nozzle welds are below the RIS 2014-11 fluence threshold of 1 x 10^{17} n/cm^2 (E > 1.0 MeV) below which the effects of embrittlement do NOT need to be considered.

Table 5.2-8 (Page 1 of 2)

Summary of Adjusted Reference Temperature (ARTs) for 54 EFPY[(a)]

Material Description	Method Used To Calculate the CF[(b)]	54 EFPY ART	
		1/4T ART (°F)	3/4T ART (°F)
Intermediate Shell Plate B9004-1	Position 1.1	148.7	139.2
Intermediate Shell Plate B9004-2	Position 1.1	124.0	115.3
	Position 2.1	157.4	142.9
Lower Shell Plate B9005-1	Position 1.1	131.1	119.0
Lower Shell Plate B9005-2	Position 1.1	126.5	116.1
Vessel Beltline Welds[(c)]	Position 1.1	63.0	46.8
	Position 2.1	13.5	5.9
Upper Shell Plate B9003-1	Position 1.1	144.1	124.1
Upper Shell Plate B9003-2	Position 1.1	148.8	130.5
Upper Shell Plate B9003-3	Position 1.1	144.1	124.1
Upper Shell Longitudinal Weld Seams 101-122 A, B, and C (Heat # 51912 (3490))	Position 1.1	47.4	14.9
Upper Shell Longitudinal Weld Seam 101-122 A, B, and C (Heat # 51912 (3536))	Position 1.1	27.4	-5.1
Upper Shell Longitudinal Weld Seams 101-122 A, B, and C (Heat # EAIB)	Position 1.1	66.2	57.9
Upper Shell Longitudinal Weld Seams 101-122 A, B, and C (Heat # IAGA)	Position 1.1	24.2	6.1
Upper Shell Longitudinal Weld Seams 101-122 A, B, and C (Heat # BOHB)	Position 1.1	111.2	85.2
Upper Shell Longitudinal Weld Seams 101-122 A, B, and C (Heat # BAOED)	Position 1.1	-14.3	-26.2
Intermediate to Upper Shell Girth Weld Seam 103-121 (Heat # 4P5174 (1122))	Position 1.1	86.6	57.4
Intermediate to Upper Shell Girth Weld Seam 103-121 (Heat # 51922 (3489))	Position 1.1	45.2	19.2
Intermediate to Upper Shell Girth Weld Seam 103-121 (Heat # AAGC)	Position 1.1	-15.8	-33.9
Intermediate to Upper Shell Girth Weld Seam 103-121 (Heat # KOIB)	Position 1.1	-5.8	-23.9

Table 5.2-8 (Page 2 of 2)
Summary of Adjusted Reference Temperature (ARTs) for 54 EFPY[a]

Material Description	Method Used To Calculate the CF[b]	54 EFPY ART
Inlet Nozzle B9011-1	Position 1.1	82.4[d]
Inlet Nozzle B9011-2	Position 1.1	110.7[d]
Inlet Nozzle B9011-3	Position 1.1	97.9[d]
Inlet Nozzle Weld Seams 105-121 A, B, & C (Heat # 4P5174 (1122))	Position 1.1	-14.6[d]
Inlet Nozzle Weld Seams 105-121 A, B, & C (Heat # LOHB)	Position 1.1	-48.1[d]
Inlet Nozzle Weld Seams 105-121 A, B, & C (Heat # HABJC)	Position 1.1	-62.2[d]
Inlet Nozzle Weld Seams 105-121 A, B, & C (Heat # BABBD)	Position 1.1	-62.2[d]
Inlet Nozzle Weld Seams 105-121 A, B, & C (Heat # FABGC)	Position 1.1	-68.1[d]
Inlet Nozzle Weld Seams 105-121 A, B, & C (Heat # EOBC)	Position 1.1	-52.2[d]
Inlet Nozzle Weld Seams 105-121 A, B, & C (Heat # FAAFC)	Position 1.1	-32.4[d]
Inlet Nozzle Weld Seams 105-121 A, B, & C (Heat # CCJC)	Position 1.1	-52.2[d]
Inlet Nozzle Weld Seams 105-121 A, B, & C (Heat # FAGB)	Position 1.1	-22.2[d]
Inlet Nozzle Weld Seams 105-121 A, B, & C (Heat # BAOED)	Position 1.1	-42.2[d]

Notes:

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

(b) Regulatory Guide 1.99, Revision 2.

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

(d) For conservatism, the Inlet Nozzles and associated welds are evaluated at the maximum fluence through the reactor vessel; thus, the ART values at the 1/4T and 3/4T locations are equal.

Table 5.2-9 (Page 1 of 1)

Calculation of the Limiting Adjusted Reference Temperatures (ARTs) for 54 EFPY[(a)]

PARAMETER	VALUES	
Operating Time	54 EFPY	
Material – Intermediate Shell Plate	B9004-2	B9004-2
Location	1/4T	3/4T
Chemistry Factor, CF (°F)	61.7	61.7
Fluence, (f), (10^{19} n/cm^2)[(b)]	3.92	1.52
Fluence Factor, FF	1.352	1.116
ΔRT_{NDT} = CF x FF(°F)	83.4	68.9
Initial RT_{NDT}, I(°F)	40	40
Margin, M(°F)	34	34
ART, per Regulatory Guide 1.99, Revision 2	157.4	142.9

Notes:

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

(b) Fluence (f), is based upon f_{surf} (10^{19} n/cm^2, E $>$ 1.0 MeV) = 6.28 at 54 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 ($x10^{19}$ 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	- - -	6.28	1.445	40.5	60	58.5	0	17	34	152.5
Intermediate Shell Plate	B9004-2	- - -	6.28	1.445	37	40	53.5	0	17	34	127.5
→ Using non-credible surveillance data[g]			6.28	1.445	61.7	40	89.1	0	17	34	163.1
Lower Shell Plate	B9005-1	- - -	6.30	1.445	51	28	73.7	0	17	34	135.7
Lower Shell Plate	B9005-2	- - -	6.30	1.445	44	33	63.6	0	17	34	130.6
Intermediate to Lower Shell Girth Weld	101-171	83642	6.27	1.444	34.4	-30	49.7	0	24.8	49.7	69.4
→ Using credible surveillance data[g]			6.27	1.444	16.1	-30	23.3	0	11.6	23.3	16.5
Intermediate Shell Longitudinal Welds	101-124 A&B	83642	1.83	1.166	34.4	-30	40.1	0	20.1	40.1	50.2
→ Using credible surveillance data[g]			1.83	1.166	16.1	-30	18.8	0	9.4	18.8	7.5
Lower Shell Longitudinal Welds	101-142 A&B	83642	1.86	1.170	34.4	-30	40.2	0	20.1	40.2	50.5
→ Using credible surveillance data[g]			1.86	1.170	16.1	-30	18.8	0	9.4	18.8	7.7

Notes:
(a) Data obtained from Table D-1 of Reference 14.
(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 D of Reference 12.

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}© (°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	0.463	0.786	91.0	50	71.5	0	17	34	155.5
	B9003-2	B4431-2	0.463	0.786	83.0	60	65.2	0	17	34	159.2
	B9003-3	A9406-2	0.463	0.786	91.0	50	71.5	0	17	34	155.5
Upper Shell Longitudinal Welds	101-122A 101-122B 101-122C	51912 (3490)	0.463	0.786	73.71	-50	57.9	0	28	56	63.9
		51912 (3536)	0.463	0.786	73.71	-70	57.9	0	28	56	43.9
		EAIB	0.463	0.786	27.0	10[d]	21.2	17	10.6	40.1	71.3
		IAGA	0.463	0.786	41.0	-30	32.2	0	16.1	32.2	34.4
		BOHB	0.463	0.786	68.0	10[d]	53.4	17	26.7	63.3	126.7
		BAOED	0.463	0.786	27.0	-50	21.2	0	10.6	21.2	-7.6
Upper to Intermediate Shell Girth Weld	103-121	4P5174	0.463	0.786	122.0	-50	95.8	0	28	56.0	101.8
		51922	0.463	0.786	68.0	-56[d]	53.4	17	26.7	63.3	60.7
		AAGC	0.463	0.786	41.0	-70	32.2	0	16.1	32.2	-5.6
		KOIB	0.463	0.786	41.0	-60	32.2	0	16.1	32.2	4.4
Inlet Nozzles	B9011-1	2V2436-01-002	0.0153	0.145	77.0	60[h]	11.2	0	5.6	11.2	82.4
	B9011-2	2V2437-02-001	0.0153	0.145	96.0	60[d][h]	13.9	17	7.0	36.7	110.7
	B9011-3	2V2445-02-003	0.0153	0.145	96.0	70[h]	13.9	0	7.0	13.9	97.9

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	0.0153	0.145	122.0	-50	17.7	0	8.9	17.7	-14.6
		LOHB	0.0153	0.145	41.0	-60	6.0	0	3.0	6.0	-48.1
		HABJC	0.0153	0.145	27.0	-70	3.9	0	2.0	3.9	-62.2
		BABBD	0.0153	0.145	27.0	-70	3.9	0	2.0	3.9	-62.2
		FABGC	0.0153	0.145	41.0	-80	6.0	0	3.0	6.0	-68.1
		EOBC	0.0153	0.145	27.0	-60	3.9	0	2.0	3.9	-52.2
		FAAFC	0.0153	0.145	95.0	-60	13.8	0	6.9	13.8	-32.4
		CCJC	0.0153	0.145	27.0	-60	3.9	0	2.0	3.9	-52.2
		FAGB	0.0153	0.145	27.0	-30	3.9	0	2.0	3.9	-22.2
		BAOED	0.0153	0.145	27.0	-50	3.9	0	2.0	3.9	-42.2

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 D-1 of Reference 14.

(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.